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U.S. Nuclear
Regulatory
Commission

1985
Annual
Report

NRC

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June 30, 1986

The President
The White House
Washington, D.C. 20500

Dear Mr. President:

This Annual Report for 1985 of the United States Nuclear Regulatory Commission is forwarded for your transmittal to the Congress, as required by Section 307(c) of the Energy Reorganization Act of 1974.

The report is devoted mainly to coverage of events and activities occurring in fiscal year 1985, with additional treatment of events after that period where circumstances warranted.

Respectfully
Nunzio J Palladino
Nunzio J. Palladino
Chairman

NRC

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NUREG-0690, *1979 NRC Annual Report*, published March 1980
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NUREG-0920, *1981 NRC Annual Report*, published June 1982
NUREG-0998, *1982 NRC Annual Report*, published June 1983
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NUREG-1145, Vol. 1, *1984 NRC Annual Report*, published June 1985

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NRC Annual Report

Statutory Reporting Requirements

ENERGY REORGANIZATION ACT OF 1974, AS AMENDED

Section 307(c) directs the Commission to include in its Annual Report statements and descriptions concerning:

"...the short-range and long-range goals, priorities, and plans of the Commission as they relate to the benefits, costs, and risks of nuclear power." (See Chapter 1 for overall policy and planning guidance. Specific goals concerning nuclear power reactors are also discussed in Chapters 2 and 3; operating experience and the evaluation thereof in Chapter 4; fuel cycle concerns in Chapter 5; safeguards in Chapter 6; waste management in Chapter 7; inspection, enforcement and emergency preparedness in Chapter 8; relations with the States in Chapter 9; international relations and nuclear nonproliferation in Chapter 10; and nuclear regulatory research in Chapter 11.)

"...The Commission's activities and findings in the following areas—

"(1) insuring the safe design of nuclear power plants and other licensed facilities. . . ." (For reactors, see Chapter 2,3 and 11; for materials facilities, devices and transportation packaging, see Chapters 5 and 11; for waste facilities, see Chapters 7 and 11.)

"(2) investigating abnormal occurrences and defects in nuclear power plants and other licensed facilities. . . ." (See Chapters 2, 3 and 4.)

"(3) safeguarding special nuclear materials at all stages of the nuclear fuel cycle. . . ." (See Chapters 6, 10 and 11.)

"(4) investigating suspected, attempted, or actual thefts of special nuclear materials in the licensed sector and developing contingency plans for dealing with such incidents. . . ." (See Chapters 6, 8 and 11.)

"(5) insuring the safe, permanent disposal of high-level radioactive wastes through the licensing of nuclear activities and facilities." (See Chapters 7 and 11.)

"(6) protecting the public against the hazards of low-level radioactive emissions from licensed nuclear activities and facilities. . . ." (See Chapters 2, 5 and 7.)

Section 205 requires development of "a long term plan for projects for the development of new or improved safety systems for nuclear power plants" and an annual updating of the plan. (See Chapter 11.)

Section 209 requires the Commission to include in each Annual Report a chapter describing the status of the NRC's domestic safeguards program. (See Chapter 6.)

Section 210 directs the Commission to submit "a plan providing for the specification and analysis of unresolved safety issues relating to nuclear reactors," and to include progress reports in the Annual Report thereafter concerning corrective actions. (See Chapter 2.)

NUCLEAR NONPROLIFERATION ACT OF 1978

Section 602 requires annual reports by the Commission and the Department of Energy to "include views and recommendations regarding the policies and actions of the United States to prevent proliferation which are the statutory responsibility of those agencies. . . ." (See Chapter 10.)

ATOMIC ENERGY ACT OF 1954, AS AMENDED

Section 170(i) directs the Commission to report annually on indemnity actions implementing the Price-Anderson Act which provides a system to pay public liability claims in the event of a nuclear incident. (See Chapter 9.)

PUBLIC LAW 96-295

Section 303 directs the Commission to report annually a statement of "(1) the direct and indirect costs to the Commission for the issuance of any license or permit and for the inspection of any facility; and (2) the fees paid to the Commission for the issuance of any license or permit and for the inspection of any facility." (See Chapter 13.)

PUBLIC LAW 97-415

Section 10(c) requires that the "Commission include as a separate chapter a description of the collaborative efforts...by the Commission and the Department of Energy with respect to the decontamination, repair or rehabilitation of facilities at Three Mile Island Unit 2." (See Chapter 3.)

1985 Highlights/1986 Planning

CHAPTER 1

This is the 11th annual report of the U.S. Nuclear Regulatory Commission (NRC). The NRC was created by enactment of the Energy Reorganization Act of 1974 as an independent agency of the Federal Government. The five Commissioners are nominated by the President and confirmed by the U.S. Senate. The Chairman of the Commission is appointed by the President from among the Commissioners confirmed.

The mission of the NRC is to assure that non-military uses of nuclear materials in the United States—as in the operation of nuclear power plants or in medical, industrial or research applications—are carried out with proper regard and provision for the protection of public health and safety, of the environment, and of the national security; for the safeguarding of nuclear materials and facilities from theft and sabotage; and for safe transport and disposal of nuclear materials and wastes. The NRC accomplishes its purposes through the licensing of nuclear reactor operations and other possession and use of nuclear materials, the issuance of rules and regulations governing licensed activities, and inspection and enforcement actions.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1985 (October 1984 through September 1985) within the NRC or involving the NRC. The report is prepared in compliance with Section 307(c) of the Energy Reorganization Act of 1974, which requires that an annual report be submitted to the President for transmittal to the Congress. Other statutory reporting requirements related to the annual report are set forth on the preceding page.

This chapter deals with salient agency decisions and actions and with some of the noteworthy events—from among those covered in detail within the body of the report—which took place during the report period. The report period is fiscal year 1985, i.e., October 1, 1984 to September 30, 1985 (some coverage is given in the report, as warranted, to events occurring in the last quarter of the calendar year 1985). The section below entitled "Policy and Planning Guidance for 1986" is based upon the policy guidance document drawn up yearly by the Commission and distributed throughout the NRC staff.

Changes Within Commission and Senior Staff

The following changes occurred on the Commission and at senior staff level during the report period:

In July 1985, Commissioner Thomas M. Roberts was reappointed to the Commission for a second term.

In January 1985, James M. Taylor was appointed Director, Office of Inspection and Enforcement, succeeding Richard C. DeYoung.

In February 1985, J. Nelson Grace was appointed Regional Administrator of the Region II Office, Atlanta, Georgia, succeeding James P. O'Reilly.

In July 1985, Ronald M. Scroggins was appointed Controller/Director, Office of Resource Management, succeeding Learned W. Barry.

Noteworthy Events of 1985

Following are some of the more significant or potentially significant decisions and actions taken by the Nuclear Regulatory Commission during the report period.

Power Reactor Regulation. During fiscal year 1985, the NRC issued low-power operating licenses (permitting fuel loading and/or operation up to 5 percent of rated power) to 10 utilities, and also granted full-power operating licenses to 10 utilities. (Eight of the 10 utilities received both low-power and full-power licenses for the same facility. See Table 2 in Chapter 2.) These actions brought the total of power reactors licensed to operate in the United States to 96. Licensing actions of all kinds taken during the report period numbered 2,949, with about 4,000 actions still pending. Because of the steadily increasing number of operating reactors coming on line and the absence of any applications for construction permits, a major shift of the workload in the Office of Nuclear Reactor Regulation (NRR) has taken place in recent years. Responding to this reality, and to the expectation that the regulation of reactor operations and safety technology concerns will DOMINATE NRR activities for the foreseeable future, the Office carried out a reorganization of staff and resource management during fiscal year 1985 (see "Improving the Licensing Process," in Chapter 2). At the damaged Three Mile Island Unit 2 facility in Pennsylvania, the cleanup continued throughout the report period. Removal of the reactor vessel plenum assembly in May 1985 permitted access to the core itself, so that the safe removal and storage of the reactor fuel could begin. (See Chapter 3.)

Inspection and Enforcement. Over 3,100 inspections of operating power reactors were carried out in fiscal year 1985, and another 1,200 inspections of facilities under construction were performed. More than 2,000 inspections of nuclear materials and over 200 fuel facilities inspections were also conducted. The NRC monitored about 70 of the full-scale emergency preparedness exercised that are required annually. In the enforcement area, there were 90 civil penalty actions taken during the report period, and 18 enforcement orders were issued.

An NRC inspector examines welds on a reactor plant containment wall. Nearly one-third of the NRC's resources are committed to its inspection program and the task of verifying the safety of licensed nuclear activities. Most inspections are carried out by personnel assigned to NRC Regional Offices and on-site at nuclear facilities. (See Chapter 8.)

A total of 109 information notices were issued during the report year, including 10 updates of notices previously issued (See Chapter 8.)

Safeguards inspections during the fiscal year totaled 474, of which 258 involved power reactors. (See Chapter 6.)

Changes in Backfit Policy. Backfitting is a process involving either plant-specific changes due to individual actions during routine licensing or inspection activities, or generic changes, those applicable to a number of plants. In April 1984, the Commission approved staff internal policy guidance to govern the agency's management of plant-specific backfitting. In December 1984, the Commission proposed to amend its regulation controlling both plant-specific and generic backfitting of commercial power reactors, 10 CFR 50.109.

In September 1985, the Commission issued the amended 10 CFR 50.109 regulation and several other conforming regulations which collectively now provide the solid foundation governing all agency backfit activities. The current rule defines backfitting as the modification of or addition to systems, structures, components or design of a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision of the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position after certain licensing milestones have been achieved. The rule applies to new plant applications, plants already approved for construction, operating plants, and standardized design approvals.

The rule generally controls backfits by stating that the Commission shall require the backfitting of a facility only when it determines, based on an analysis as described in the rule, that there is a substantial increase in the overall protection of the public health and safety or the common defense or security to be derived from the backfit and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection.

For those backfits which the NRC will seek to impose, a systematic and documented analysis will be prepared addressing a number of relevant and material factors, including:

(1) Statement of the specific objectives that the proposed backfit is designed to achieve;

(2) General description of the activity that would be required by the licensee or applicant in order to complete the backfit;

(3) Potential change in the risk to the public from the accidental offsite release of radioactive material;

(4) Potential impact on radiological exposure of facility employees;

(5) Installation and continuing costs associated with the backfit, including the costs of facility downtime or the cost of construction delay;

(6) The potential impact of changes in plant or organization complexity, including the relationship to proposed and existing regulatory requirements;

(7) The estimated resource burden on the NRC associated with the proposed backfit and the availability of such resources;

(8) The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed backfit;

(9) Whether the proposed backfit is interim or final and if interim, the justification for imposing the proposed backfit on an interim basis.

The NRC may, however, impose backfits without meeting the standard of the rule or completing the backfit analysis described above provided that either: (i) That a modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or (ii) That an immediately effective regulatory action is necessary to ensure that the facility poses no undue risk to the public health and safety.

All requirements, including backfit requirements, proposed by the NRC staff related to one or more classes of reactors must be reviewed by the Committee to Review Generic Requirements (CRGR). The Committee seeks to eliminate unnecessary demands on licensees by ensuring that the need for a new requirement can be demonstrated by those proposing it. (See the 1982 NRC Annual Report, pp. 1-3, for full description of CRGR's structure and review process. Through its review, the CRGR seeks assurance that a proposed requirement is (1) necessary for the public health and safety, (2) likely to result in a net safety improvement, and (3) have an impact on the public, industry and government which is consistent with and justified by the urgency of the need for the safety improvement to be realized. Following its review, the CRGR recommends to the Executive Director for Operations (EDO) that the proposed requirement be approved, disapproved, modified or conditioned in some way. It also makes recommendations as to the method and scheduling of implementation. The EDO considers CRGR recommendations, as well as those of cognizant NRC offices in deciding whether a requirement shall be imposed. From its inception in November 1981 through September 1985, the CRGR has held 81 meetings and considered a total of 128 separate issues.

The plant-specific backfit policy, revised and reissued in November 1985 to assure conformance with the new backfit regulation, was reviewed with principal staff members at seminars in each regional office and the major headquarters offices during the year. In addition, seminars have been completed in all regions explaining the basic principles involved in preparing benefit/cost analyses for those backfit issues that are amenable to such quantitative analyses. A computer based Plant-Specific Backfit System provides a common agency data base for monitoring of the progress made in resolving each backfit issue.

Safety Goals. In March 1983, the Commission published (48 *FR* 10772) a "Policy Statement on Safety Goals for the Operation of Nuclear Power Plans." This policy statement was a revision of one previously issued for public comment (47 *FR* 7023, February 17, 1982). The revised policy statement contained preliminary safety goals and preliminary numerical design objectives that were intended to be consistent with the goals. The Commission also published a staff Evaluation Plan that was to be used during a two-year period of evaluation of the

Safety Goal Policy Statement. A Safety Goal Evaluation Steering Group was appointed by the EDO to guide the staff's evaluation activities during the two-year period.

The conclusions and supporting technical evaluations by the Steering Group were presented to the Commission and released to the public in mid-1985. The basic conclusions of the report are as follows:

(1) The use of Safety Goals can strengthen decisionmaking by adding more objectivity and predictability to the regulatory process. The safety goals will be valuable as a regulatory yardstick against which a wide range of regulatory issues can be measured.

(2) The basic structure of the Safety Goals is sound, and they are not in need of radical revision.

(3) The core melt guideline should be given nearly as much weight as the individual and societal mortality risk design objectives in order to enhance the defense-in-depth safety philosophy and to be more useful in practice as a screening criterion.

(4) Probabilistic Risk Assessment (PRA) and Safety Goals should not be used within a framework of strict acceptance or nonacceptance criteria for regulatory decisionmaking.

(5) The staff expects to make substantial use of Safety Goal comparisons to augment, but not supplant, traditional safety review methods for making regulatory decisions.

As of October 1985, the staff was studying the results of the technical evaluations and a proposed revised Policy Statement. An overall agency proposal to the Commission regarding a Policy Statement and a proposed implementation plan for the utilization of safety goals in the regulatory process is expected to be submitted to the Commission in early 1986.

Adequate Used Fuel Storage. The Commission amended its regulations during the report period to implement certain provisions of the Nuclear Waste Policy Act of 1982, which requires that the Commission establish criteria and procedures for determining when the owner of a nuclear plant cannot reasonably be expected to provide continued storage of used fuel at the reactor site or at another site owned by the licensee. When this is the case, the Federal Government must provide interim storage. The NRC criteria require, among other things, that a utility requesting such a determination show evidence that it has diligently pursued alternative storage possibilities.

New Legislative Proposal. A revised version of the legislation proposed by the NRC to the Congress in 1983 was submitted during fiscal year 1985. Under the proposed law, the NRC would be empowered to grant a construction permit and an operating license in a single licensing procedure, providing for early resolution of licensing issues and thus reducing the uncertainties attending the construction of nuclear facilities. The new approach envisions the use of approved standardized nuclear plant designs matched to pre-approved plant sites.

New Emergency Operations Center. Construction of the new NRC Operations Center was completed by the end of January 1985. Final acceptance testing of the facility took place in February, and the operational changeover occurred on February 27, 1985. Among the operational improvements provided by the new center are: (1) quantity, quality, and dedication of space, (2) a better telephone system, (3) sophisticated audio/video displays, and (4) a dedicated computer system.

Policy on Severe Accidents. In August 1985, the Commission issued a policy statement (Commissioner Asselstine dissenting) on severe accidents involving existing plants and also with respect to future plant designs. Among the salient conclusions in the statement was the judgment that, on the basis of currently available information, existing nuclear power plants in the United States pose no undue risk to the public health and safety and that there was no present basis for immediate action on a generic rulemaking or other regulatory change because of the risk of severe accidents, i.e., accidents involving severe damage to the reactor core. Ongoing NRC safety programs—including Unresolved Safety Issues and generic safety issues resolution, the "source term" research program, operational data evaluation, and others—were deemed sufficient, together with the various NRC inspections programs, to give reasonable assurance that public health and safety are being protected. Significant safety information from any source which might challenge the conclusion that no undue risk exists would be dealt with under the backfit policy (see above) or other existing procedures. Criteria for assessing the acceptability of new designs are set forth in the statement.

Progress on Consolidation. Since its inception, the NRC has sought a remedy to the broad dispersion of its Headquarters staff in various venues in and around the Washington, D.C. area. The Congress, the Government Accounting Office, the Office of Management and Budget (OMB), and a number of study commissions have stressed that its multiple office locations have impeded NRC's ability to accomplish its mission. Past efforts to bring about a consolidation of staff offices have fallen short of success.

At the close of the report period, the Administrator of the Government Services Administration (GSA) had indicated a preference for pursuing NRC consolidation through the purchase of one or more buildings, and the Commission had determined that the purchase of an acceptable building(s) in suburban Montgomery County (Md.) would be acceptable. The GSA was pursuing a purchase option on a specific building at that time. The preferred site includes a single building, which could house approximately one-half of the Headquarters' operation. That would mean that six buildings currently assigned to NRC in the District of Columbia, and in Silver Spring and Bethesda in Maryland, could be relinquished. With the purchase of the building and site, sufficient land would acquired to construct a second building adequate to house the remainder of the agency.

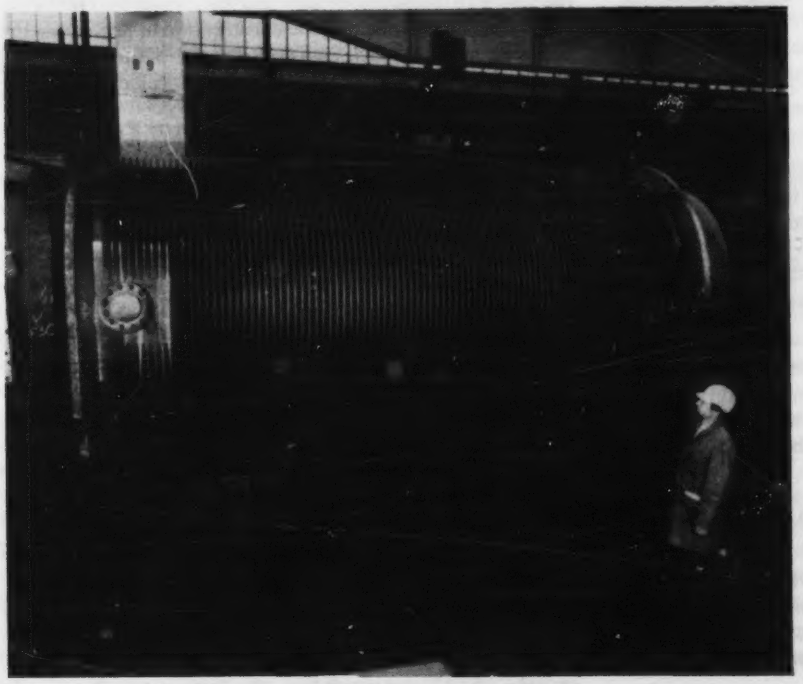
The GSA has presented the proposed purchase plan to OMB for their review as to funding availability. Contingent on OMB approval, GSA will commence final negotiations and proceed to purchase the building. Initial occupancy could begin as early as July 1986.

Policy and Planning Guidance for 1986

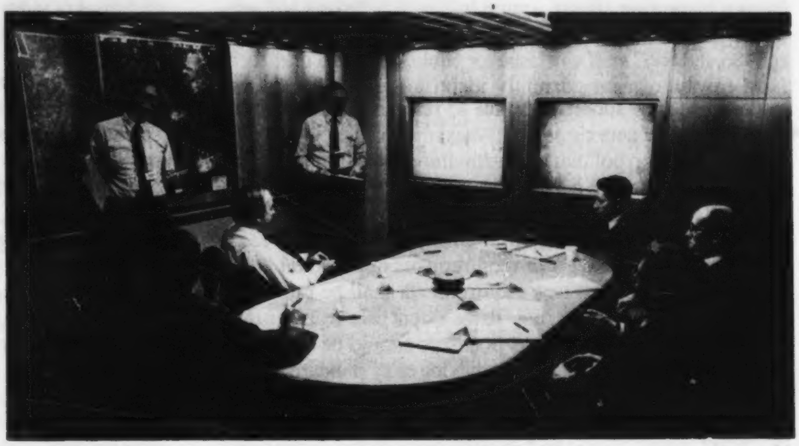
Each year the Commission publishes policy and planning guidance in a formal document which sets forth the principles and objectives underlying the regulatory philosophy of the Commission; articulates its major policies and identifies its goals; and gives guidance to the entire NRC staff in developing plans and programs, establishing priorities and allocating resources.

In 1986, the Commission is planning to publish a Five-Year Plan which will identify the program and resources needed to attain the Commission's strategic goals. The plan will be reviewed and revised annually to reflect changes in the regulatory environment and budget realities. The policy and planning guidance for 1986 will serve as the policy foundation in formulating the Five-Year Plan.

Regulatory Philosophy. The Commission's fundamental mission is to regulate those who commercially produce, possess and/or use nuclear materials so that the public health and safety, the common defense and security, and the environment are protected. The Commission recognizes that its actions can affect the nation's energy supply, to which nuclear energy is



Used fuel storage remained a central regulatory issue in 1985, and the NRC staff reviewed several designs for dry cask storage. Shown here is the "Castor V" cask; a topical report on this design was submitted by General Nuclear Systems, Inc., in partnership with Gesellschaft fur Nuklear Service of West Germany and Chem-Nuclear Corp. of the United States, in January 1985. The cask would hold 21 pressurized water reactor fuel assemblies. (See Chapter 7.)



Shown here is part of the NRC's new emergency Operations Center in Bethesda, Md. The Executive Team, headed by NRC Chairman Nunzio J. Palladino, third from right, can call up information related to plant and public safety on two video monitors on the back wall. Maps of all nuclear power plant sites are available on wall panels at left.

a significant and growing contributor. The way the Commission carries out its fundamental mission must be consistent with and complementary to the determination of the Congress that the safe use of nuclear energy for peaceful purposes, particularly in the production of electricity, is a legitimate and important national goal. With the understanding that safety considerations are paramount, the Commission continues to pursue predictability, stability and discipline in the licensing process. New requirements will be imposed on existing licensees only in accordance with the Commission's backfit rule (see above) and supporting policies, and the NRC will also encourage the nuclear industry to develop standardized plant designs.

The Commission has a statutory obligation to license a nuclear project when the NRC staff's comprehensive review of plans therefor has satisfied the Commission that the facility can be built and operated safely. At the same time, the NRC's review process should provide an accessible avenue for the expression of public concerns and an adequate response to those concerns. The Commission recognizes that the nuclear industry, once heavily engaged in construction, is now almost exclusively occupied with the operation of existing nuclear power plants. Accordingly, the NRC intends to shift its regulatory emphasis away from prescriptive requirements toward more general, performance-based requirements, and to encourage industry initiatives to improve safety. During 1986, major objectives of the NRC are to achieve technical resolution of unresolved safety issues and appropriate generic issues; develop and implement agency policy on severe accidents; support confirmatory research; continue to advocate licensing reform legislation; pursue a disciplined approach to backfitting; complete the reassessment of radioactive source terms and, where appropriate, implement revised source terms and corresponding regulations; issue and implement agency policy on advanced reactor concepts and designs; develop processes necessary to license new power plants, including standardized and advanced reactors,to reactivate deferred construction project and to extend plant operating licenses; monitor implementation of the Low Level Radioactive Waste Policy Amendment Act of 1985; and continue to urge industry improvements in construction quality assurance.

The following is a condensed statement of basic themes and mission areas set out in the NRC Policy and Planning Guidance for 1986.

- **Assuring Safe Operation of Facilities.** It is a fundamental tasks of the NRC to assure that existing nuclear reactors and those coming on-line operate safety. The agency's highest priority will be given to seeing that operating facilities maintain adequate levels of protection to public health and safety, and that reactors are adequately designed, built and tested prior to operation. While the industry has prime responsibility for safety of design, construction and operation of nuclear plants, the NRC should stress the development of commercial reactor operating expertise within the agency through training, hiring and close communication with industrial experts. The analysis of operational data, risk-based analysis, systematic assessment of licensee performance, and monitoring of performance indicators will enable the NRC to assess licensee management, assure that unresolved safety issues are promptly resolved, and evaluate major reactor safety systems under postulated accident conditions. The formulation of a severe accident policy and the early resolution of outstanding technical issues are major facets of this program.

PLANNING GUIDANCE: The staff's inspection of operating reactors should continue to focus on the plant operations of licensees. Priority attention will be given to licensees with low performance ratings, and the staff should continue to closely monitor the first two years of operation of new plants coming on line, particularly those of licensees who have little or no prior experience with nuclear plants. A timely report to the Commission will be made whenever additional regulatory attention is necessary.

The NRC should emphasize to licensees and industry that theirs is the responsibility to assure the quality of vendor-supplied equipment and services. The NRC staff should, by its own inspection efforts, assure that both licensees and vendor organizations are meeting their responsibilities. By the end of 1986, the NRC staff will issue for public comment draft technical resolutions for currently identified unresolved safety issues. The staff should continue to review and approve the addition of new generic safety issues in accordance with current Commission policies, and expeditiously to implement the Commission's severe accident policy.

- **New Regulatory Requirements.** The NRC must be sensitive to the large number of requirements imposed on licensees and take care that new requirements shall be processed in accord with the backfit rule. To the extent practicable, safety issues which affect numerous licensees should be addressed in the context of Commission rulemaking or by standard orders, as opposed to case-by-case review. Licensees should be allowed the flexibility to select the most cost-effective ways of meeting NRC safety objectives, particularly for plant-specific requirements. Licensing responsibilities shall be carried out by the NRC efficiently, in order to provide for the timely review and implementation of changes necessary to assure safe plant operation and allow for timely responses to the public under 10 CFR Part 2.206.

PLANNING GUIDANCE: The Committee to Review Generic Requirements (CRGR) shall continue to review and make recommendations to the Executive Director for Operations on proposed generic requirements. The staff should manage backfitting for reactors under construction or in operation in accordance with Commission regulations. Where practical, the results of cost-benefit analysis should be used in evaluating new requirements. Compliance schedules should reflect the importance of the requirement to safety or safeguards, as well as the licensee's ability to complete the necessary engineering, evaluation and design. The Commission will consider alternate regulatory approaches which recognize the contributions of industry initiatives to the extent that they are effect and consistent with NRC responsibilities.

- **Standardized Plant Design.** The Commission endorses regulatory actions which will encourage industry to pursue standardization of power reactor designs. The advantages of such standardized designs benefit public health and safety by concentrating industry resource on particular approaches to design problems, stimulating standardized programs of construction practice and quality assurance, fostering improved maintenance and operation, and permitting more efficient and effective licensing and inspection processes.

PLANNING GUIDANCE: During 1986, the staff should develop revised procedures to review and license new standardized nuclear plant designs and to review and pre-approve potential plant sites. During the remainder of the 1980s, the NRC should direct its efforts toward encouraging industry to proceed with standardization.

- **Investigations and Enforcement.** The NRC should adhere to fair and effective investigative and enforcement activities to assure that licensees will correct performance deficiencies where necessary and maintain an adequate level of protection of the public health and safety, safeguards and the environment. Inspections and investigations on which enforcement actions are based should identify the basic reasons for the deficiencies and violations. The Office of Investigations and the Office of Inspector and Auditor shall investigate significant allegations of wrongdoing. When initial collection of evidence indicates that criminal violations of the Atomic Energy Act may be involved, appropriate referral will be made to the Department of Justice.

PLANNING GUIDANCE: The Office of Investigations, in coordination with the Executive Director for Operations, should expeditiously complete development of criteria for

The Commission's Policy and Planning Guidance for 1986 stressed again that reactors, both those operating and coming on line, must be designed, built, tested and operated so as to ensure that public health and safety are protected. The Shearon Harris plant at New Hill, N.C., shown here under construction, is scheduled for completion in 1986.

initiating and terminating investigations. Information resulting from such investigations and which is of potential safety significance should be referred to the appropriate NRC office immediately. The Commission has established an Ad Hoc Advisory Committee for Review of the Enforcement Policy and will consider the Committee's recommendations by June 1986.

- **Reactivation of Projects.** The NRC staff should establish necessary procedures for the resumption of the licensing process for deferred power plants. Requests for an operating license renewal are to be anticipated and will require advanced planning and analysis. The Commission intends to continue development of policies and criteria for operating license extensions to assure that industry's efforts are focused on primary regulatory concerns.

PLANNING GUIDANCE: In view of the number of plants that have been postponed in the midst of construction, the staff will consider the legal and technical ramifications involved in reactivating a project after construction and licensing processes have stopped. The staff should, within appropriate jurisdictional boundaries, propose policy guidance for such projects by the end of 1986. Beginning in fiscal year 1987, the NRC should be prepared for possible requests to restart construction on deferred plants and should develop policy guidance and licensing criteria for operating license extensions.

- **Timely Licensing of Facilities.** The NRC intends that its regulatory processes be efficient and cost effective, without compromising safety, safeguards or environmental requirements. The Commission reaffirms its statement of policy of May 1981 on licensing proceedings which urged Licensing Boards to assure a more efficient conduct of hearings. The NRC should continue to support the need for licensing reform in a manner that will not detract from public participation and disclosure of information related to radioactive risks to the public.

PLANNING GUIDANCE: Staff reviews and public hearings for nuclear facilities should be completed on a schedule that assures the licensing process will not unnecessarily delay reactor startup.

- **Safety Goals.** The Commission has developed preliminary safety goals and related safety guidance and will continue to evaluate their future regulatory potential. The Commission continues to believe in emergency backup systems, containment integrity and emergency planning as essential parts of the defense-in-depth philosophy. Assumptions made in planning for nuclear emergencies should be based on the best available scientific data.

PLANNING GUIDANCE: In 1986, the staff will provide the Commission with recommendations on proposed safety goals resulting from the two-year evaluation period. Work shall proceed toward refining the use of probabilistic risk assessment techniques to implement Commission policy on safety goals, and toward developing an integrated program for the collection, analysis and distribution of data needed for risk assessment.

- **Advanced Reactors.** The NRC will maintain its capability to respond to innovative and advanced reactor designs that might be presented for Commission review. The licensing and regulation of advanced reactor concepts shall be in accord with requirements within the framework of the Advanced Reactor Policy statement.

PLANNING GUIDANCE: The staff should identify necessary changes to nuclear power plant design criteria and regulations to accommodate advanced reactors, evaluating the safety characteristics of new reactor types as such concepts evolve and are presented to the NRC.

- **Protecting Nuclear Material and Facilities.** Safeguards regulation, as an integral and ongoing element of the Commission's responsibility, should be conducted with the same defense-in-depth approach the agency employs in safety regulation. Implementation of safeguards requirements shall be conducive to the safe operation of a facility and commensurate with threat levels as approved by the Commission. Emphasis should be given to performance-based requirements rather than to prescriptive requirements in order to allow licensees to select the most cost-effective ways to satisfy NRC requirements.

The proliferation of nuclear explosives technology poses a threat to the security interests of the United States. Hence, the NRC will carefully discharge its licensing responsibilities to ensure that necessary controls are applied to the import and export of nuclear materials, equipment and facilities. The agency will encourage to the extent practicable the conversion

The year 1985 brought new threats to the security of nuclear facilities in many parts of the world. NRC planning guidance for the year focused staff attention on the importance of safeguarding licensed facilities in the U.S. Protective measures include TV surveillance, shielded access control points, two-way remote communications and internal controls for gatelocks, and many others. Fenced protected areas, such as that above, feature full illumination at night and infra-red intrusion-detection equipment.

of non-power reactors to the use of low-enriched uranium, in accordance with NRC rules.

PLANNING GUIDANCE (Domestic): The staff should continue to evaluate domestic safeguards events—within both the defense and the regulated community—in order to determine the nature of threats to the environment. In addition to assuring that safeguards plans are in place at operating facilities, the staff will conduct its independent assessment as to whether safeguards regulations adequately support NRC objectives. An annual report shall be made to the Commission detailing results of the previous year's assessments, with recommendations for necessary regulatory changes. The staff will implement the rule converting non-power reactors to low-enriched uranium fuel, and propose for Commission review any additional physical security measures necessary at research reactors.

PLANNING GUIDANCE (International): The NRC will continue to meet its commitments for the implementation of international safeguards at U.S. licensing facilities and to work with the Executive Branch as the nation pursues improvements in international safeguards.

- **Nuclear Materials.** Byproduct, source and special nuclear materials must receive regulatory attention from the NRC in order to achieve the highest level of control over potential hazards to the public and to users of these materials. The Commission will continue to encourage standardization of material licensing reviews and consistency in application of performance-based requirements. The Commission intends to pursue regulatory efforts to improve radiography safety and minimize medical misuse of radioactive materials.

The transportation of nuclear and radioactive materials is an important part of NRC regulatory responsibility.

PLANNING GUIDANCE: By mid-1986, regulations to consolidate and streamline the safety requirements associated with medical use of byproduct materials and well-logging should be promulgated, along with associated regulatory guidance, standard review plans and inspection procedures. Efforts to improve radiography safety, through the establishment of performance-based requirements, as well as more effective training and inspection, should be completed by July 1986. The staff should assure that NRC responsibilities in regulating the transportation of special nuclear materials and radioactive substances are coordinated with other Federal agencies for an integrated program to protect the public, common defense and security, and the environment.

- **Managing Nuclear Waste.** The NRC High-Level Waste Management Program is critical to the success of an urgent national priority. The NRC will provide the necessary licensing and regulatory oversight for the Executive Branch's program as required by the Nuclear Waste Policy Act of 1982 (NWPA), the Atomic Energy Act, the Energy Reorganization Act, the National Environmental Policy Act, and the Commission's regulations. Expeditious and safe cleanup of the damaged reactor at Three Mile Island Unit 2 (Pa.) is a continuing NRC safety priority. While direct responsibility for cleanup activities rests with the licensee, the NRC will continue to provide oversight and, if necessary, direction to ensure a safe decontamination of the facility and a safe and timely removal of radioactive materials.

PLANNING GUIDANCE: The NRC will continue its technical program to support the development of licensing criteria and evaluation methods, and the early identification and resolution of technical and quality assurance issues, in keeping with the requirements of the Nuclear Waste Policy Act. The staff should review existing and proposed regulations addressed by the NWPA and make conforming changes as necessary. Procedures for documenting agreements between the NRC and Department of Energy (DOE) staffs on the resolution of technical issues in advance of license review should be formalized. The NRC should review in a timely manner utility proposals for adding spent fuel storage capacity, consistent with safety and legal requirements and without unnecessarily affecting reactor operations. The NRC must also be prepared to conduct licensing reviews specified by the NWPA for any limited Federal interim storage capacity of spent fuel which may be proposed by the DOE. Insofar as resources permit, the staff shall monitor the implementation of the Low Level Radioactive Waste Policy Amendment Act of 1985 and shall keep the Commission informed of any problems requiring Commission action.

The NRC should continue to monitor closely the safe removal and disposition of nuclear wastes from the cleanup of Three Mile Island Unit 2. The NRC should also the DOE in developing plans for the safe and timely off-site disposition of the damaged core.

- **Research Program.** The NRC research program, which is essential to many aspects of the NRC mission, should provide the technical basis for rulemaking and regulatory decisions; support licensing and inspection; assess the feasibility and effectiveness of safety improvements; and increase understanding of phenomena relative to regulatory actions for which analytical methods are needed. The NRC will continue to maintain a long-range research plan, consistent with the agency's mandate and NRC's Five-Year Plan and directed toward areas of importance to the licensing and inspection processes. In particular, the Commission has decided to proceed expeditiously with further characterization of radioactive source terms.

PLANNING GUIDANCE: Research resources should be allocated to support a balanced program between research to reinforce or revise the current regulatory base and conceptual research intended to improve reactor safety, waste management and other licensed activities. Joint research programs with industry groups, other Government agencies and foreign groups should be pursued whenever possible. The severe accident research program must provide timely information for the Commission's decisionmaking process.

CHAPTER 2

Reactor Regulation

The Office of Nuclear Reactor Regulation (NRR) is responsible for regulating operating nuclear reactors, for reviewing applications for construction permits and operating licenses for new reactors and for issuing such permits and licenses after consideration by the Advisory Committee on Reactor Safeguards, Atomic Safety and Licensing Boards and Appeal Boards, and the Commission. These functions require resolution of generic and specific issues with regard to safety, the environment, and antitrust matters.

This chapter summarizes NRR activities during fiscal year 1984, under the following headings: Status of Licensing/License Amendments, Improving the Process, Human Factors, Unresolved Safety Issues, Safety Reviews, Protecting the Environment, and Antitrust Activities. Also included in this chapter is a section on the activities of the Advisory Committee on Reactor Safeguards.

Status of Licensing

Applications for Operating Licenses, Construction Permits or Manufacturing Licenses

NRC received no new applications for operating licenses, construction permits or manufacturing licenses during fiscal year 1985. Ten utilities were issued 11 Low-Power Licenses (permitting fuel load at 0 percent power or low-power operation at 5 percent power) during fiscal year 1985. In addition, Full-Power Operating Licenses were issued to 10 utilities.

Table 1 summarizes NRR activity in power reactor licensing during fiscal year 1985. Table 2 provides greater detail concerning the licenses issued.

The staff is reviewing applications for operating licenses for all 33 units under construction; the schedules for these reviews are consistent with the projected plant completion dates. Some of these units have been delayed indefinitely and eventually may be cancelled. Two units owned by the Public Service Company of Indiana—Marble Hill Nuclear Generating Station, Units 1 and 2—were cancelled during fiscal year 1985.

Licensing Actions for Operating Power Reactors

At the end of fiscal year 1985, 96 power reactors were licensed to operate in the United States. Both routine post-licensing activities and unexpected events at these facilities can result in a need for licensing actions. Routine post-licensing activities affecting the operation of already licensed reactors include license amendment requests, public hearings, requests for exemption from regulations, new regulations that require backfit modifications to operating reactors, orders for modification of a license, new generic activities, and review of information supplied by a licensee for the resolution of technical issues. In addition, unexpected events create a large number of licensing actions. These two sources have produced an inventory of approximately 4,000 pending licensing actions. Table 3 summarizes activity related to operating licenses during fiscal 1985.

Licensing Actions for Non-power Reactors

On October 1, 1984, 65 non-power reactors licensed for operation by the NRC were in use for research, training and testing. Nine applications for operating license renewals were being reviewed or were otherwise pending. During fiscal year 1985, the staff issued seven license renewals for operation and one for fuel possession only. The staff received applications for one construction permit and four operating license renewals; the construction permit and two of the four renewals were issued during the year. This is the first construction permit of any kind issued by the NRC since 1979. The reviews of the three remaining renewal applications are scheduled to be completed during the first quarter of fiscal year 1986 and the staff expects to receive two additional operating license renewal applications during fiscal year 1986. The backlog of operating license renewals has been completely eliminated, permitting reviews to begin as soon as the applications are received.

One renewal application was contested by an intervenor. However, the licensee and the intervenor were able to reach an agreement and completion of the license renewal is expected to be completed during the first quarter of fiscal year 1986. The agreement provides for the license to be renewed for possession only rather than operation.

Table 4 summarizes fiscal year 1985 licensing actions for non-power reactors.

Table 1. Power Reactor Licensing — FY 1985

Low-Power Operating Licenses issued	11 (10 utilities)
Full-Power Operating Licenses issued	11 (10 utilities)
Safety Evaluation Reports issued	3
Draft Environmental Impact Statements issued	4
Final Environmental Impact Statements issued	7
Operating Licenses under Review	33
Applications Cancelled	2 (Marble Hill 1 and 2)
Construction Permits issued	0
Construction Permits under Review	0
Manufacturing Licenses issued	0
Manufacturing Licenses under Review	0

Special Cases

Comanche Peak. As construction of the Comanche Peak Steam Electric Station Unit 1 (Tex.) neared completion, a number of issues remained to be resolved before the staff could make a licensing decision. These issues were quite complex and involved several NRC offices. To ensure the overall coordination/integration of these issues and to maintain a schedule consistent with hearing and licensing decision needs, the Executive Director for Operations directed NRR to manage and coordinate all NRC offices in activities related to a licensing decision on Comanche Peak. First, NRR identified the issues to be resolved before hearing and licensing decisions could be made, and the regulatory actions to be taken in the areas of licensing, inspections, allegations, and investigations. Next, NRR developed a plan for resolving the issues and then implemented the necessary regulatory actions.

Of major import were numerous adverse allegations, most of which concerned construction adequacy and quality assurance. To investigate these allegations, NRR assembled a Technical Review Team (TRT) on site. The TRT included more than 50 technical experts from the NRC (NRR and Regions), national laboratories, and consulting organizations. The TRT spent four months on the site investigating these allegations. They documented their findings in five Supplemental Safety Evaluation Reports.

In addition, numerous other concerns about the design and construction of the plant evolved through contentions before the NRC's Atomic Safety Licensing Board (ASLB) and the Comanche Peak Independent Assessment Program review conducted by Cygna Energy Services.

In response to these concerns, the applicant has submitted portions of a program plan for their resolution. The NRC staff has completed its initial reviwe of the programmatic aspects of the material submitted to date. The plan is comprehensive and provides a structure capable of addressing all relevant issues, existing and future. The applicant also has committed to programs intended to demonstrate the adequacy of plant design and construction. The staff will complete its evaluation when the applicant submits its final plan modified in response to staff comments, and provides a comprehensive quality assurance (QA) program for activities identified in the plan. Refion IV presonnel will then audit implementation of the plan, assisted by NRR, IE, and consultants. Because of plant inspections and modifications, fuel loading planned for the furst quarter of 1986 will be delayed. No new fuel load date has been announced.

Indian Point Units 2 and 3. The Union of Concerned Scientists filed a petition with the Commission in September, 1979, requesting that Indian Point Units 2 and 3 (N.Y.) be shut down. In February, 1980, the Director of NRR denied the request but imposed certain interim requirements by Order while an nrc Task Force determined what changes should be made to reduce the probability and/or consequences of a severe reactor accident. In addition, in February 1981, the Commission initiated a special investigatory proceeding in which one issue was the risk reduction resulting from the requirements imposed by the Order. After hearing the testimony on this issue, the Licensing Board concluded, in October 1983, that the NRR Order resulted in a small benefit not amenable to quantification. After reviewing the Licensing Board's recommendation, the Commission concluded, in May 1985, that, because the risk reduction benefits of the Order were not substantial, the Director's Order should be rescinded unless it was required to fulfill generic requirements applicable to similar types of reac-

Table 2. Licenses Issued in FY 1985 for Operation of Nuclear Power Plants

Applicant	*Facility*	*Fuel Load License*	*Low-Power License*	*Full-Power License*	*Location*
Union Electric	Callaway			10/18/84	10 Miles SE of Fulton, Mo.
Philadelphia Electric	Limerick 1		10/26/84	8/8/85	35 Miles NW of Philadelphia, Pa.
Commonwealth Edison	Byron 1		10/31/84	2/14/85	17 Miles SW of Rockford, Ill.
Pacific Gas & Electric	Diablo Canyon 1			11/2/84	12 Miles WSW of San Luis Obispo, Cal.
Duke Power	Catawba 1		12/6/84	1/17/85	6 Miles NNW of Rock Hill, S.C.
Long Island Lighting	Shoreham	12/7/84	7/3/85		Brookhaven, N.Y.
Louisiana Power & Light	Waterford 3		12/18/84	3/16/85	20 Miles W of New Orleans, La.
Arizona Public Service	Palo Verde 1		12/31/84	6/1/85	36 Miles West of Phoenix, Ariz.
Kansas Gas & Electric	Wolf Creek		3/11/85	6/4/85	3.5 Miles NE of Burlington, Kans.
Detroit Edison	Fermi 2		3/20/85	7/15/85	Laguna Beach, Mich.
Pacific Gas & Electric	Diablo Canyon 2		4/26/85	8/26/85	12 Miles WSW of San Luis Obispo, Cal.
Gulf States Utilities	River Bend		8/29/85		24 Miles NNW of Baton Rouge, La.

tors, or was required to meet other license requirements for the Indian Point units. Accordingly, pursuant to the Commissioners Decision CLI-85-06, a Recision of Order was issued on July 5, 1985.

Davis-Besse. On June 9, 1985, the Davis-Besse plant near Toledo, Ohio, experienced a loss-of-feedwater event, followed by a number of equipment failures. This was one of the most significant events at an operating nuclear power plant since the Three Mile Island (Pa.) accident in 1979. An NRC fact-finding team immediately went to the site. After the team had published its findings in NUREG-1154, NRR instituted a program of short and long term improvements to address the plant-specific and generic implications of the event. Numerous inspections and technical reviews were undertaken. The results of these reviews and the safety significance of the event are discussed in greater detail under "Safety Reviews," later in this chapter. (See also Chapter 4.) The staff held special meetings with the licensee (Toledo Edison Company) and the Commission, and testified at Congressional subcommittee hearings in June and October.

At the close of the report period, the Davis-Besse plant was in cold shutdown. Fact-finding was nearly complete, and corrective actions and system modifications were under way. The staff is continuing its review of the fact-finding team's investigations. The staff is also reviewing the licensee's detailed program of actions that must be done before Davis-Besse can be restarted. (See discussion of "feed-and-bleed" capabilities at the plant, under "Safety Reviews," later in this chapter.)

Byron Station Unit 1. On January 13, 1984, an Atomic Safety and Licensing Board denied authorization of an operating license for Byron Station Unit 1 (Ill.) because of inadequacies in quality assurance. The Appeal Board subsequently remanded the case to the Licensing Board for further evidence, particularly with respect to results of a reinspection program

Table 3. Operating Reactor Licensing Actions — FY 1985

Power Reactors Licensed to Operate	96
Licensing Actions under review, 10/1/84	3800 (approximate)
Licensing Actions issued during FY 1985	2949
Licensing Actions under review, 9/30/85	4000 (approximate)

that had not been before the board during the earlier hearing.

The Licensing Board issued its supplemental decision on October 16, 1984, setting aside its earlier denial and authorizing the Director of NRR to issue an operating license. The board also made the required findings on safety and environmental matters not at issue in the hearing. Byron 1 received an operating license on October 31, 1984, authorizing it to load fuel and conduct testing at up to 5 percent of full power. Authorization for full-power operation was issued on February 14, 1985.

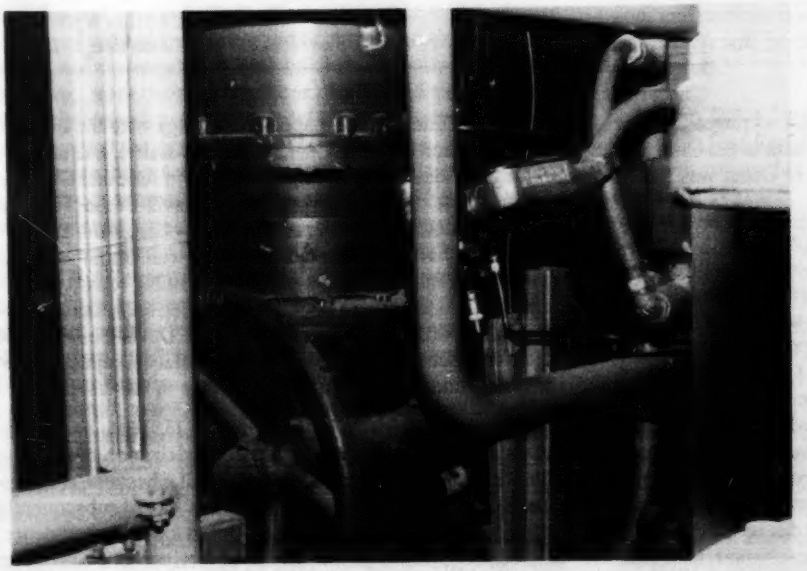
San Onofre Unit 1 Restart. San Onofre Unit 1 (Cal.) was shut down on February 27, 1982, for steam generator inspections and plant modifications. Because of new seismic analyses submitted by the licensee in May 1982, the staff became concerned about whether the plant met the original licensing basis for earthquake protection. At that time, the licensee, Southern California Edison Company, committed to upgrading the plant to meet the staff's reevaluation of the design-basis earthquake (0.67 g) before restarting the unit. The NRC confirmed this commitment by an Order dated August 11, 1982.

The licensee originally committed to first upgrade those systems necessary for safe shutdown and then to implement the other modifications. As the modifications proceeded, however, the licensee reassessed this commitment and in late 1983, proposed to revise it. The new commitment stipulated that before the plant was restarted, the licensee would complete all modifications needed to enable the plant to achieve and maintain a hot standby condition after a 0.67 g earthquake. After reviewing the licensee's proposal, the staff concluded that implementation of the proposal would provide reasonable assurance that the plant can be safely shut down following an earthquake of 0.67 g.

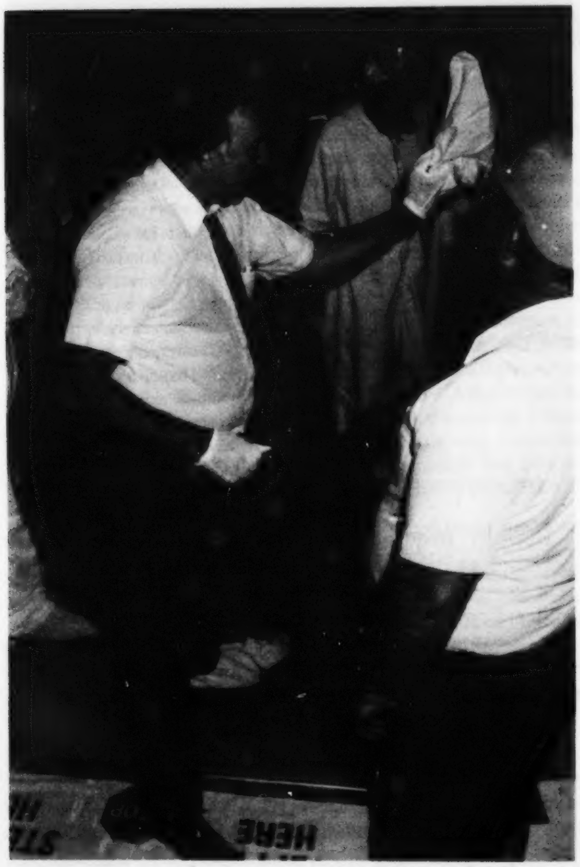
On November 21, 1984, the NRC issued a Contingent Recision of Suspension of the August 11, 1982 Order and authorized the licensee to restart the plant on the condition that the remaining modifications are completed before startup, after the refueling outage scheduled to begin November 30, 1985.

Shoreham. During fiscal year 1985, three Atomic Safety and Licensing Boards (ASLB) held hearings on issues related to the Shoreham nuclear facility on Long Island, N.Y.

The licensee had requested an exemption from the Commission's general design criteria governing plant electric power systems. This request was approved by an ASLB and partially upheld by the Appeal Board and the Commission. As a result,



This visibly damaged turbine bypass valve was one of several equipment failures involved in a loss of main and auxiliary feedwater at the Davis-Besse nuclear power plant (Ohio) on June 9, 1985. The plant's two steam generators were without feedwater for about 12 minutes.



The event of June 9, 1985 at the Davis-Besse facility was one of the most significant since the Three Mile Island accident in 1979. NRC officials were on the scene in force from first notification. Dr. Harold Denton, Director of Nuclear Reactor Regulation, is shown here removing protective gear following one of many inspection tours of the plant.

a fuel loading and cold criticality license was granted on December 7, 1984. Shoreham achieved initial criticality on February 15, 1985.

On June 14, 1985, another ASLB ruled that installation of the Transamerica Delaval, Inc., emergency diesel generators complied with the Commission's regulations. A new license, authorizing operation at up to 5 percent of rated power, was issued on July 3, 1985. Low-power testing continued to the end of fiscal year 1985.

On April 17 and August 26, 1985, a third ASLB issued partial initial decisions regarding off-site emergency planning. The Board ruled that, although the licensee's plan generally conforms to the Commission's regulations, the lack of state and county participation in executing the emergency plan precludes the issuance of a full-power operating license. These decisions have been appealed to the Appeal Board. Until the emergency planning issues are resolved, Shoreham will be limited to operation below 5 percent of rated power.

Diablo Canyon. The Commission granted a full-power license for Diablo Canyon Unit 1 (Cal.) in November 1984, a low-power license for Unit 2 in April 1985, and a full-power license for Unit 2 in August 1985. These licensing actions entailed the resolution of more than 1,700 allegations that had been submitted to the Commission through June 1985 by former employees of the licensee, Pacific Gas and Electric Company (PG&E), or its contractors and subcontractors. The staff gave its evaluation of the allegations in Supplement 26 and 28 to the Diablo Canyon Safety Evaluation Report (SER). A final supplement on this matter is scheduled for late calendar year 1985.

A separate report, SER Supplement 30, addressed matters related to Unit 2 piping and pipe supports that arose from allegations and were based on a Unit 1 license condition. A request by the joint Intervenors for a hearing on the Unit 2 piping and pipe supports was denied by the Atomic Safety and Licensing Appeal Board.

Before reaching its full-power license decision in August 1984, the Commission had raised the issue of the need for additional consideration of possible complicating effects of an earthquake on the Diablo Canyon emergency plan. This issue was subsequently brought before the U.S. Court of Appeals for the District of Columbia where a panel of that Court affirmed the Commission decision that no specific consideration of this matter was required. However, in May 1985 the full Court ordered a rehearing en banc for early October 1985.

The staff approved the licensee's long term seismic program plan developed in response to a license condition in the Unit 1 full-power license. The program will reevaluate the seismic design basis for the Diablo Canyon plant by considering information and methodologies that have become available since 1978. The program is expected to be completed by mid-1988.

Three Mile Island, Unit 1 Restart. On February 13, 1985, the Commission issued Order CLI 85-2 that stated, in part, that hearings in progress were to be completed, but no new or additional hearings were necessary to reach a restart decision regarding Unit 1 at the Three Mile Island nuclear facility (Pa.). On May 29, 1985, the Commission issued Order CLI 85-9 removing the "immediate effectiveness" provision of the July 1979 shutdown order for TMI-1. The Commission's Order was appealed and the U.S. Court of Appeals issued the first of several temporary stays on the effectiveness of Commission Order CLI 85-9. Subsequently, on October 2, 1985, the Supreme Court vacated the stay, and the Commission's Order lifting the suspension became effective. Later the same day, the Director of NRR authorized the operation of TMI-1.

The licensee began startup activities in accordance with an NRC-approved power ascension schedule. The startup protocol will take approximately 99 days to complete and the Regional Administrator must give his approval before activities can proceed past specified hold points. Enhanced inspection coverage began early in the morning on October 3, 1985. At 1:30 pm on October 3, 1985, TMI-1 became critical for the first time since 1979.

In related hearing matters, a partial initial decision on the remanded issue of TMI-1 operator training was issued May 3,

Table 4. Licensing Actions for Non-Power Reactors — FY 1985

Non-power reactor operating licenses, 10/1/84	65
OL renewals issued for operation	9
OL renewals issued for possession only	1
CP issued	1
Licenses amended to possession only	2
Licenses terminated	4
Orders issued to decommission/dismantle	3
Facilities in process of decommission/dismantle	4
OL renewals under review, 10/1/85	3

1985, in favor of the licensee. The issue in contention in this hearing embraces all licensed operator training at TMI-1. Originally the hearing had a much smaller scope, concerning only one aspect of operator training, but the licensee chose to adduce its entire licensed operator program as its defense, and the hearing was expanded to adapt to the licensee's presentation.

On August 19, 1985, a partial initial decision in favor of the licensee was issued in the "Dieckamp mailgram" hearing. On May 9, 1979, Mr. Herman Dieckamp, President and Chief Executive Officer of General Public Utilities Corporation (GPU), sent a mailgram to Congressman Morris Udall (D. Ariz.) concerning licensee knowledge of pressure "spikes" and their relation to core damage during the TMI-2 accident. The purpose of the hearing was to determine if the mailgram knowingly contained material false information. Both decisions are now under appeal.

Finally, by Order dated September 6, 1985, the Commission initiated a hearing to determine if Mr. Charles Husted should have supervisory responsibilities for training licensed and non-licensed personnel. Mr. Husted is an employee of GPU. As a result of the restart hearings, the Commission decided to bar Mr. Husted from training non-licensed operators at TMI-1. GPU already has an agreement with the Commonwealth of Pennsylvania that prevents Mr. Husted from training licensed personnel. The Commission agreed to allow Mr. Husted an opportunity for a hearing on this decision, which could directly affect his employment.

Tennessee Valley Authority (TVA). TVA holds operating licenses for five nuclear power units, three at Browns Ferry near Decatur, Ala., and two at Sequoyah near Chattanooga, Tenn. TVA also has four units under construction, two at Watts Bar near Spring City, Tenn., and two at Bellefonte near Scottsboro, Ala. One Watts Bar unit is ready for licensing; the other is about 75 percent complete. Completion of construction for one unit at Bellefonte has been delayed to 1993 and for the other to 1995.

However, frequent problems at TVA nuclear plants, both operating and under construction, have culminated in the current shutdown of all TVA operating units and lengthy delays in licensing Watts Bar Unit 1.

To develop an integrated agency approach for dealing with TVA, the NRC Executive Director for Operations has designated a Senior Executive Board. On September 17, 1985, the staff issued a letter in accordance with 10 CFR 501.54(f) advising TVA of information needed for the staff to reach licensing decisions. TVA is to provide this information 90 days before a request for a fuel load license for Watts Bar Unit 1 or 60 days before restart of any operating unit. A Commission paper was developed presenting this approach (SECY-85-231), and the staff met with the Commission on September 12, 1985 to discuss the status of all TVA plants.

Browns Ferry. Despite a Regulatory Performance Improvement Program designed to address earlier weaknesses, licensee performance at Browns Ferry (Ala.) during fiscal year 1985 has not shown significant improvement. During the fiscal year, the staff imposed five civil penalties against Browns Ferry, with fines totaling $462,500. All three units are shut down, and the licensee has stated that the units will remain down until the licensee is satisfied it has proper management capabilities and can demonstrate compliance with all NRC requirements.

The licensee proposes to implement an Operational Readiness Review plan to gauge readiness to restart. However, although the staff requested a copy of this plan in its September 17, 1985 letter, TVA had not yet submitted it for staff review by the close of the report period. The letter also outlines other issues TVA will be required to address. These include design control, environmental qualification of equipment, updating of the integrated schedule, and changes in site management.

Sequoyah. Operations were suspended at the Sequoyah plant on August 21, 1985. TVA shut down Units 1 and 2 because a review by WESTEC Services, a private consulting firm, and by TVA staff found that available documentation could not sub-

THE LICENSING PROCESS

Obtaining an NRC construction permit—or a limited work authorization (see discussion below) prior to a decision on issuance of a construction permit—is the first objective of a utility or other company seeking to operate a nuclear power reactor or other nuclear facility under NRC licensing authority. The process is set in motion with the filing and acceptance of the application, generally comprising 10 or more large volumes of material covering both safety and environmental factors, in accordance with NRC requirements and guidance. The second phase consists of safety, environmental, safeguards and antitrust reviews undertaken by the NRC staff. Third, a safety review is conducted by the independent Advisory Committee on Reactor Safeguards (ACRS); this review is required by law. Fourth, a mandatory public hearing is conducted by a three-member Atomic Safety and Licensing Board (ASLB), which then makes an initial decision as to whether the permit should be granted. This decision is subject to appeal to an Atomic Safety and Licensing Appeal Board (ASLAB) and could ultimately go to the Commissioners for final NRC decision. The law provides for appeal beyond the Commission in the Federal courts.

As soon an initial application is accepted, or "docketed," by the NRC, a notice of that fact is published in the Federal Register, and copies of the application are furnished to appropriate State and local authorities and to a local public document room (LPDR) established in the vicinity of the proposed site, as well as to the NRC public document room in Washington, D.C. At the same time, a notice of a public hearing is published in the Federal Register and local newspapers which provides 30 days for members of the public to petition to intervene in the proceeding. Such petitions are entertained and adjudicated by the ASLB appointed to the case, with rights of appeal by the petitioner to the ASLAB.

The NRC staff's safety, safeguards, environmental and antitrust reviews proceed in parallel. With the guidance of the Standard Format (Regulatory Guide 1.70), the applicant for a construction permit lays out the proposed nuclear plant design in a Preliminary Safety Analysis Report (PSAR). If and when this report has been made sufficiently complete to warrant review, the application is docketed and NRC staff evaluations begin. Even prior to submission of the report, NRC staff conducts a substantive review and inspection of the applicant's quality assurance program covering design and procurement. The safety review is performed by NRC staff in accordance with the Standard Review Plan for Light-Water-Cooled Reactors, initially published in 1975 and updated periodically. This plan sets forth the acceptance criteria used in evaluating the various systems, components and structures important to safety and in assessing the proposed site; it also describes the procedures to be used in performing the safety review.

The NRC staff examines the applicant's PSAR to determine whether the plant design is safe and consistent with NRC rules and regulations; whether valid methods of calculation were employed and accurately carried out; whether the applicant has conducted his analysis and evaluation in sufficient depth and breadth to support staff approval with respect to safety. When the staff is satisfied that the acceptance criteria of the Standard Review Plan have been met by the applicant's preliminary report, a Safety Evaluation Report is prepared by the staff which summarizes the results of its review regarding the anticipated effects of the proposed facility on public health and safety.

Following publication of the staff Safety Evaluation Report, the ACRS completes its review and meets with staff and applicant. The ACRS then prepares a letter report to the Chairman of the NRC presenting the results of its independent evaluation and recommending whether or not a construction permit should be issued. The staff issues a supplement to the Safety Evaluation Report incorporating any changes or actions adopted as a result of ACRS recommendations. A public hearing can then be held, generally in a community near the proposed facility site, on safety aspects of the licensing decision.

In appropriate cases, the NRC may grant a Limited Work Authorization to an applicant in advance of the final decision on the construction permit in order to allow certain work to begin at the site, saving as much as seven months time. The authorization will not be given, however, until NRC staff has completed environmental impact and site suitability reviews and the appointed ASLB has conducted a hearing on environmental impact and site suitability with a favorable finding. To realize the desired saving of time, the applicant must submit the environmental portion of the application early.

The environmental review begins with an assessment of the acceptability of the applicant's Environmental Report (ER). If the ER is judged sufficiently complete to warrant review, it is docketed, and an analysis of the consequences to the environment of the construction and operation of the proposed facility at the proposed site is begun. Upon completion of this analysis, a Draft Environmental Statement is published and distributed with specific requests for review and comment by Federal, State and local agencies, other interested parties and members of the public. All of their comments are then taken into account in the preparation of a Final Environmental Statement. Both the draft and the final statements are made available to the public at the time of respective publication. During this same period, the NRC is conducting an analysis and preparing a report on site suitability aspects of the proposed licensing action. Upon completion of these activities, a public hearing—with the appointed ASLB presiding—may be held on environmental and site suitability issues related to the proposed licensing action. (Or a single hearing on both safety and environmental matters may be held, if that is indicated.)

The antitrust reviews of license applications are carried out by the NRC and the Attorney General in advance of, or concurrent with, other licensing reviews. If an antitrust hearing is required, it is held separately from those on safety and environmental aspects.

About two or three years before construction of a plant is scheduled to be completed, the applicant files an application for an operating license. A process similar to that for the construction permit is followed. The application is filed, the NRC staff and the ACRS review it, a Safety Evaluation Report and an updated Environmental Statement are issued. A public hearing is not mandatory at this stage, but one may be held if requested by affected members of the public or at the initiative of the Commission. Each license for operation of a nuclear reactor contains technical specifications which set forth the particular safety and environmental protection measures to be imposed upon the facility and the conditions that must be met for the facility to operate.

Once licensed, a nuclear facility remains under NRC surveillance and undergoes periodic inspections throughout its operating life. In cases where the NRC finds that substantial, additional protection is necessary for the public health and safety or the common defense and security, the NRC may required "backfitting" of a licensed plant, i.e., the addition, elimination or modification of structures, systems or components of the facility.

stantiate the environmental qualification of equipment. TVA stated that the units would remain shut down until TVA can be sure that safety equipment will operate properly in an emergency situation.

Unit 1 is now in an early refueling outage, and neither unit is expected to start up until November 1985.

On September 17, 1985, NRC directed TVA to furnish information on Sequoyah 60 days before the restart of either unit. The information requested includes an operational readiness plan, an analysis of cable tray supports in the diesel generator building, a discussion of any generic implications on plant design, and a long term program to ensure continued compliance with regulations.

Watts Bar. The need to complete modifications for fire protection and problems with surveillance instruction delayed licensing from October 1984 until April 1985 at the Watts Bar facility. About that time, the NRC informed TVA of safety concerns about Watts Bar Unit 1 expressed anonymously to NRC by TVA employees.

To identify and resolve these concerns, TVA hired the Quality Technology Company to interview all TVA employees who worked on Watts Bar. The interviews identified more than 1,000 potential safety-related concerns that TVA is now investigating. Although the staff has not yet reviewed the TVA program, the staff has held several preliminary meetings on the topic.

The staff also has established a Dedicated Review Group to review the results of the Watts Bar Independent Design Verification Program conducted for TVA by Black and Veatch. This group also will review the concerns presented by TVA employees. Although other licensing issues must be resolved, the staff considers the resolution of employee concerns the paramount issue.

Improving the Licensing Process

Reorganization of NRR

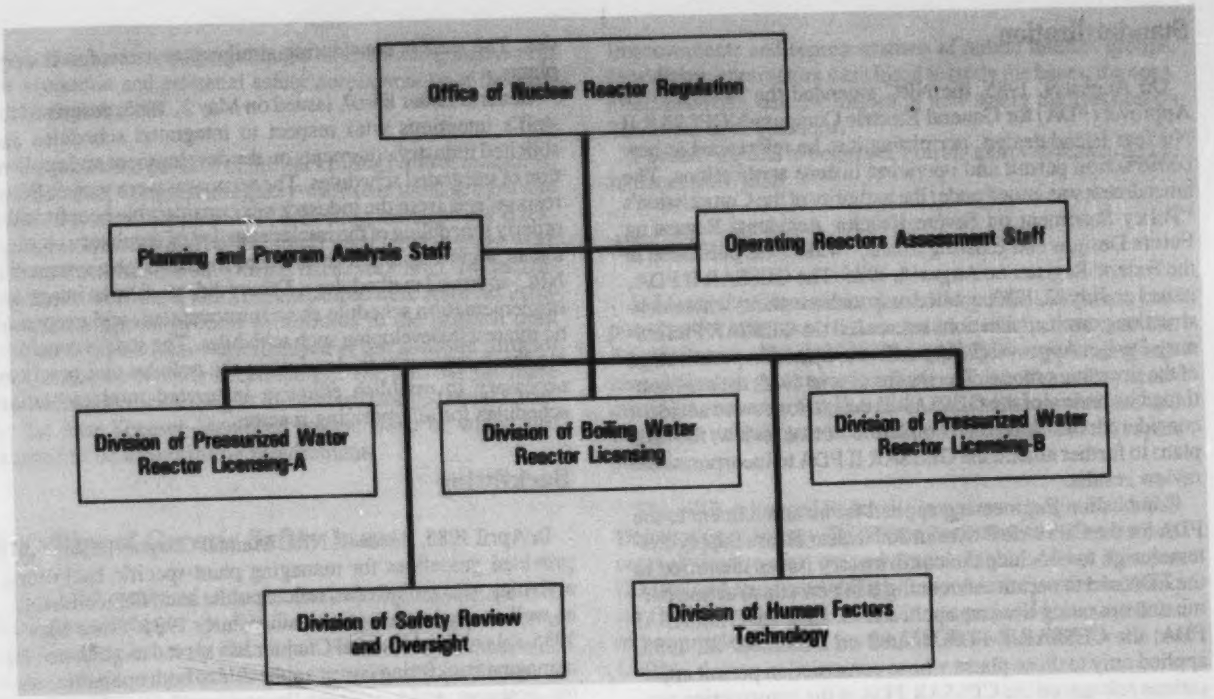
Over the last few years nuclear reactor regulation has shown two major trends. First, the bulk of NRR work has shifted from license application reviews to the regulation of operating reactors, which has created an imbalance of NRC staff workload, since some of the technical skills required for licensing reviews are only rarely needed for operating reactors work. Secondly, the nuclear industry is tending to turn to the reactor vendors to meet current and future design demands; owners and industry groups have been formed to resolve reactor problems, and standard plants are being designed by each of the vendors.

Based on these trends, a scenario for NRR's future workload may be projected. The regulation of reactor operations and safety technology activities are expected to dominate. Fiscal year 1986 resource allocation is expected to be approximately 60 percent for operating reactors, 20 percent for safety technology and the remaining 20 percent for all other responsibilities. Future reviews of license applications are likely to reflect the efficiencies of design standardization, which has been and is being vigorously advocated by the Congress and the Commission. The industry is similarly stressing standardization, with strong encouragement by the Electric Power Research Institute (EPRI). Two of the four vendors of nuclear power reactors, General Electric and Westinghouse, have already developed standard design plants which are under staff review. It is expected that new applications for construction permits are likely to reference an approved standard design. The scenario also assumes that only cost-effective safety and regulatory improvements will be made. To facilitate this, the NRC will make the regulatory process more predictable by promulgating a safety goal, by issuing well-defined backfit regulations, and by promoting other mechanisms to enhance licensing.

To accommodate this projection of future activities, NRR initiated efforts to reorganize its staff and resource management during fiscal year 1985. The basic objective of the proposed new organization is to continue timely completion of Planning and Program Guidance goals of the Commission and the Executive Director for Operations (EDO). Licensing decisions will continue to be of high priority and schedules will be maintained consistent with assurance of public health and safety and protection of the environment. Continuity of reactor reviews and safety issue resolution will be maintained. The expected benefits will include the creation of an organization that is consistent with the changing workload and with the new industry/NRC approach discussed above. Staff expertise will be better focused on reactor operations and NRR efficiency/effectiveness will be improved by separation of more forward-looking activities from "day-to-day" licensing activities. The project manager role will be enhanced and his responsibility and authority increased. This is expected to shorten lines of communication and to improve management of operating reactor licensing actions. Overhead will be reduced while increasing NRR's organizational flexibility to address major operating reactor problems without creating special task forces.

The new organization will have two Staffs and five Divisions reporting to the Director of NRR. The Planning and Program Analysis Staff will provide administrative management and coordination of the programs and resources of the Office. Their current functions will be expanded to include Technical Assistance Project Management and they will assume many of the reporting and accountability functions currently performed by the Division of Licensing. The Operating Reactors Assessment Staff (ORAS) will be based on the existing Operating Reactors Assessment Branch and will systematically assess operating experience to identify significant events for detailed review by the appropriate technical support groups. In keeping with the orientation along vendor lines, the new ORAS will divide its tasks into matters related to boiling water reactors and those related to pressurized water reactors.

There will be three Project Divisions. One Division will be dedicated to the regulation of Westinghouse-designed reactors, one to General Electric reactors, and one to Combustion Engineering and Babcock and Wilcox reactors; there will also be several other special projects covered in the last-named Divi-


Office of Nuclear Reactor Regulation
Planning and Program Analysis Staff
Operating Reactors Assessment Staff
Division of Pressurized Water Reactor Licensing-A
Division of Boiling Water Reactor Licensing
Division of Pressurized Water Reactor Licensing-B
Division of Safety Review and Oversight
Division of Human Factors Technology

sion, such as non-power reactors, the TMI-2 cleanup and the Integrated Safety Assessment Program. Each Project Division will employ the project management and technical skills required to perform all needed licensing activities for the reactors assigned. Within the divisions, plants will be segregated into Project Directorates—according to design parameters, age, power level and other criteria. For example, Project Directorate #1 in the Division of PWR Licensing-A will be responsible for the oldest plants. These will have a capacity 400-500 MWe, large dry containments, and will include all existing 2-loop designs.

The Division of Safety Review and Oversight will be the central focal point for major technical subjects. They will provide the lead for forward-looking safety issues, such as advanced reactor designs, research and standards coordination, safety goals, source term and severe accident phenomenology, unresolved safety issues (USIs) and other generic issues. They will also provide the oversight necessary to ensure consistency among the various Project Divisions in the implementation of regulatory requirements and guidance. The review of full scope Probabilistic Risk Assessments (PRAs) and NRR's efforts to refine PRA methods will take place within this Division, while a portion of the reliability and risk assessment expertise will be transferred to each of the three Project Divisions.

The Division of Human Factors Technology will assume all of the duties performed by the existing Division of Human Factors Safety, including all aspects of reactor operator licensing, the Human Factors Program Plan, activities related to plant maintenance and personnel training, and coordination of technical specifications associated with operating licenses. In order to integrate human factors information into the day-to-day work of the Project Divisions, the current human factors engineering, psychology, procedures development, and expertise regarding utility organizational structure—currently being applied to operating reactors under licensing review—will be transferred to the Project Divisions.

The chart below shows major relationships within the new NRR organization; also shown is a matrix relating the present organization functions to the proposed new organization.

Decentralization

NRC transferred responsibility for the review of about 600 licensing actions to the five Regional Offices during fiscal year 1985. This brings the total number of licensing action reviews transferred to the Regions since fiscal year 1982 to approximately 1,100. These reviews include inservice testing, emergency exercise exemption requests, organizational changes, snubber surveillance, degraded grid voltage testing, and various plant-specific issues. Regional personnel conduct technical reviews, make site visits when appropriate, and prepare safety evaluation reports for NRR. A two-year pilot program to evaluate the effectiveness of these regional reviews was completed in June 1985 with the issuance of the NRR Executive Report, which summarized the assessment of regional licensing action reviews. The general conclusion was that the program was useful and should be continued.

Standardization

On August 9, 1985, the NRC amended the Final Design Approval (FDA) for General Electric Company's GESSAR II Nuclear Island design, permitting it to be referenced in new construction permit and operating license applications. The amendment was issued under the authority of the Commission's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," which was published in the Federal Register on August 8, 1985. The GESSAR II FDA, issued on July 27, 1983, applied only to those plants whose construction permit applications referenced the GESSAR Preliminary Design Approval (PDA) at the construction permit stage of the licensing process. During fiscal year 1985, the staff continued its review of the GESSAR II design for severe accident considerations. Upon the completion of that review, the staff plans to further amend the GESSAR II FDA to incorporate the review results.

Combustion Engineering applied for an amendment to the FDA for the CESSAR-F System 80 Nuclear Steam Supply System design to conclude the confirmatory issues identified in the FDA and to permit referencing it in new construction permit and operating licenses applications. Like the GESSAR II FDA, the CESSAR-F FDA, issued on December 21, 1983, applied only to those plants whose construction permit applications referenced the CESSAR PDA at the construction permit stage of the licensing process. The staff review of Combustion Engineering's amendment request was continuing at the close of the report period, and a decision was expected in the first quarter of fiscal year 1986.

The staff also continued its review of the Westinghouse Electric Corporation application for the PDA for its RESAR-SP/90 Nuclear Power Block design, docketed on May 19, 1984. A decision on the issuance of the PDA is scheduled for fiscal year 1987. Westinghouse intends to pursue an FDA and a Design Certification for its RESAR-SP/90 design.

The NRC is continuing discussions with the Electric Power Research Institute (EPRI) on its Advanced LWR Standard Plant Program. Beginning in fiscal year 1986, EPRI expects to develop and submit for NRC review a series of "requirements documents." These documents would serve as a basis for the development by the industry of boiling water reactor and pressurized water reactor standard nuclear power plant designs.

The staff also has proposed revision to the Commission's 1978 standardization policy statement. This revision would reflect the experience the NRC has acquired in implementing the 1978 policy statement, the applicable provisions of the severe accident policy statement and of proposed standardization legislation, and the current views of the Commission and industry on standardization. The revised policy statement is expected to be issued in fiscal year 1986.

Integrated Implementation Schedules

The licenses of two operating power plants incorporate formal scheduling processes for implementing new and existing requirements with appropriate consideration of relative priorities. The staff is considering similar provisions for 11 other plants.

Generic Letter 85-07, issued on May 2, 1985, described the staff's intentions with respect to integrated schedules and solicited industry comments on the development and application of integrated schedules. The responses were varied. Some respondents from the industry saw considerable benefit in the orderly scheduling of the implementation of regulatory requirements, according to priorities established through a systematic, NRC-approved methodology. Others did not view an integrated implementation schedule as an improvement, and expressed no interest in developing such schedules. The staff is considering these responses in developing the policies and practices necessary to establish effective integrated implementation schedules for all operating reactors.

Backfitting

In April 1985, the draft NRC Manual Chapter 0514, which provided guidelines for managing plant-specific backfitting activities, was modified to reflect public and NRC comments, as well as experience gained since early 1984. Since May 1, 1985, this revised Manual Chapter has served as guidance for managing backfitting issues applicable to both operating reactor licensees and operating license applicants. Significant changes from the original draft Manual Chapter include the following: (1) a determination will be made by the staff and approved by the NRR Office Director as to whether an issue is a backfitting requirement before further action is taken, and (2) a regulatory analysis will be made and approved before a backfitting requirement is imposed on a licensee, unless prompt imposition is necessary. After the revised Manual Chapter was issued, the Office of the Executive Director for Operations conducted seminars in all the Regions and at Headquarters on proper procedures for identifying backfitting issues and for conducting the backfitting process in a manner consistent with Manual Chapter 0514.

The final backfitting rule affirmed by the Commission was published in the *Federal Register* on September 20, 1985. The revised draft Manual Chapter must now be modified to reflect the provisions of the new rule and issued in final form. The final Manual Chapter will include the following modifications: (1) information requests other than those pertaining to compliance issues will be evaluated to ensure that the burden to be imposed on the respondents is justified with respect to the potential safety significance, (2) facilities with standardized designs are referenced in the backfitting definition, and (3) a regulatory analysis will be performed after an immediate imposition of a backfitting requirement to document its safety significance (cost considerations of this analysis are to be included only insofar as they contribute to selecting among alternatives). Each Headquarters and Regional Office will develop procedures to implement the final Manual Chapter. These procedures are intended to provide a uniform and consistent approach to making backfitting decisions, to encourage effective interchange between the licensee and the staff on backfitting issues, and to outline guidance on conducting analyses of the economic and potential safety consequences of the proposed requirement.

To monitor the efficacy of the backfitting control measures, the NRC staff designed an agency-wide data management and retrieval system that provides backfitting management tracking to headquarters and regional staff via microcomputer work stations at each location. During fiscal year 1985, 24 backfitting issues were resolved. Of these, three were resolved using the interim appeal process procedures in the original draft Manual Chapter (two were resolved at the assistant director level, one at the division director level). The staff has also made determinations according to the revised draft Manual Chapter for four licensee-identified issues, none of which was deemed to be a backfitting requirement.

Priorities of Generic Safety Issues

The NRR continued to use the methodology cited in the *1982 NRC Annual Report* (p. 29) for determining the priority of generic safety issues. In December 1983, a comprehensive list of the issues subjected to this method was published in NUREG-0933, "A Prioritization of Generic Safety Issues," and is updated semiannually, with supplements in June and December. This list includes items from the TMI Action Plan (NUREG-0660) and Unresolved Safety Issues (USIs), which are discussed in detail later in this chapter. The results of the NRC's continuing effort to identify significant unresolved safety issues will be included in future Supplements to NUREG-0933.

The staff identified 38 new generic safety issues in fiscal year 1985, including 18 Human Factors Program Plan (HFPP) subtask issues. Priorities for 27 issues listed in Table 5 were established in fiscal year 1985. NRC resolved 23 issues other than USIs; these are listed in Table 6. Eight-eight generic safety issues, plus 24 HFPP issues, remain unresolved. Table 7 contains the schedules for the resolution of these issues.

Technical Specifications

NRR initiated a Technical Specification Improvement Project (TSIP) in January 1985 to develop and carry out a plan of action to ensure that Technical Specifications (which are issued as part of each power reactor operating license) are focused on important operational safety matters. TSIP will also make recommendations for reducing operational and licensing problems associated with Technical Specifications. The project staff of senior technical people, guided by a management level interoffice advisory group, completed a study and plan of action for management review. During the course of its study, the TSIP staff met with vendor and utility personnel, the Atomic Industrial Forum's (AIF) Subcommittee on Technical Specification Improvements and representatives of public interest groups. In addition, contractors were hired to study the bases, the operational impact, and the impact of new safety information on Technical Specifications.

The staff expects to begin work on an approved plan of action in fiscal year 1986.

Advanced Reactors

In March the Commission issued for public comment a proposed policy statement on the regulation of advanced reactors. Approximately 20 organizations responded, representing utilities, industry, government and the public. The policy statement was revised to accommodate the comments received and was under review by the Commission at the close of the report period.

The NRR Advanced Reactors Group is working with the U.S. Department of Energy (DOE) and its contractors to review conceptual designs for two advanced Liquid Metal Reactors (LMRs) and an advanced High Temperature Gas-Cooled Reactor (HTGR). The reviews are in progress and are scheduled to be complete in 1987 for the HTGR and in 1988 for the two LMRs.

Allegations Management System

In July 1984, NRR assumed responsibility for the NRC Allegations Management System, a resource designed to provide appropriate and expeditious response to safety-related allegations affecting nuclear power plant licensing and/or operation. This responsibility had been held in the Office of Inspection and Enforcement. The initial NRR tasks were the design, development and implementation of a new computer-based system for tracking and managing allegations. Throughout the first half of fiscal year 1985, NRR worked closely with the Office of Resource Management (ORM) and the principal offices involved in the resolution of allegations to bring the new system on line.

As of April 2, 1985, personnel in all NRC Regional Offices and major Headquarters Offices have direct, on line capability and can access the system through micro-computers. For the first time since the agency began tracking allegations and their resolution in December 1982, offices responsible for the resolution of an allegation can manage their own data base and can update this information through on-line, interactive capabilities. With further improvements, the system will be able to provide a variety of management reports.

During fiscal year 1985, NRC received about 820 allegations, and resolved about 690. At any one time, there are between 500 and 600 allegations under review. These numbers do not include allegations regarding the Comanche Peak (Tex.) facility, discussed under "Special Cases," earlier in this chapter.

Table 5. Issues Prioritized in FY 1985

Number	*Title*	*Priority*
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	COVERED IN USI A-47
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	COVERED IN II.E.6.1
55.	Failure of IE Safety-Related Switchgear Circuit Breakers	DROP
59.	Technical Specification Requirements for Plant Shutdown	COVERED IN TECHNICAL SPECIFICATION IMPROVEMENT PROGRAM
67.	Steam Generator Staff Actions	MEDIUM
81.	Potential Safety Problems Associated With Locked Doors and Barriers in Nuclear Power Plants	DROP
84.	CE PORVs	NEARLY RESOLVED
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	DROP
86.	NRC Pipe Cracking Review Group Study	NEARLY RESOLVED
87.	Failure of HPCI Steam Line Without Isolation	HIGH
91.	Transamerica Delaval Emergency Diesel Generator Main Crankshaft Failure	NEARLY RESOLVED
93.	Steam Binding of Auxiliary Feedwater Pumps	HIGH
94.	Additional Low-Temperature-Overpressure Protection For Light Water Reactors	HIGH
98.	CRD Accumulator Check Valve Leakage	DROP
99.	RCS/RHR Suction Line Interlocks on PWRs	DROP
101.	BWR Water Level Redundancy	HIGH
102.	Human Error in Events Involving Wrong Unit or Wrong Train	COVERED IN HF-02
103.	Design For Probable Maximum Precipitation	NEARLY RESOLVED
105.	Interfacing System LOCA at BWRs	HIGH
108.	BWR Suppression Pool Temperature Limits	LOW
119.	Piping Review Committee Recommendations	NEARLY RESOLVED
122.	Hydrogen Control for Large Dry Containment	HIGH
B-19	Thermal-Hydraulic Stability	NEARLY RESOLVED
B-50	Post Operating Basis Earthquake Inspection	LOW
B-59	N-1 Loop Operation in BWRs and PWRs	RESOLVED
HF-01	Human Factor Program Plan (HFPP twenty-four subtask elements)	HIGH
HF-02	Maintenance and Surveillance Program Plan (MSPP ten subtask elements)	HIGH

Note: HIGH, MEDIUM, and NEARLY RESOLVED priority issues are allocated resouces for resolution. DROP and LOW priority issues are not allocated resources for resolution.

Table 6. Generic Safety Issues Resolved in FY 1985

Number	*Title*
22	Inadvertent Boron Dilution Events
A-41	Long Term Seismic Program Requirements
B-19	Thermal-Hydraulic Stability
B-54	Ice Condenser Containments
B-58	Passive Mechanical Failures
C-11	Assessment of Failure and Reliability of Pumps and Valves
I.A.2.2	Training and Qualifications of Operating Personnel
I.A.2.6 (4)	Operator Workshops
I.A.2.7	Accreditation of Training Institutions
I.A.3.4	Licensing of Additional Operations Personnel
I.G.2	Scope of Test Program Requirement
II.B.6	Risk Reduction For Operating Reactors at Sites With High Population Densities
II.B.8	Rulemaking Proceedings on Degraded Core Accidents (a) Hydrogen Rule (b) Severe Accidents
II.C.1	Interim Reliability Evaluation Program
II.C.2	Continuation of Interim Reliability Evaluation Program
II.E.2.2	Research on Small Break LOCAs Medium and Anomalous Transients Requirement
III.A.1.3(2)	Maintain Supplies of Thyroid-Blocking Agent For Public
III.A.3.4	Nuclear Data Link
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants
III.D.2.3(2)	Discriminate Between Sites and Plants that Require Consideration of Liquid Pathway Interdiction Techniques
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction
III.D.2.3(4)	Prepare a Summary Assessment
IV.E.5	Assess Currently Operating Plants

Table 7. Generic Issues Scheduled for Resolution

A. NRR Issues

Issue Number	*Title*	*Priority*	*Schedule Resolution Date*
14	PWR Pipe Cracks	NEARLY RESOLVED	09/85
23	Reactor Coolant Pump Seal Failures	HIGH	07/85
29	Bolting Degradation or Failures in Nuclear Power Plants	HIGH	08/86
36	Loss of Service Water (Calvert Cliffs Unit 1)	NEARLY RESOLVED	05/86
48	LCO for Class 1-E Vital Instrument Buses in Operating Reactors	NEARLY RESOLVED	02/87
49	Interlocks and LCOs for Class 1-E Tie Breakers	MEDIUM	09/88
51	Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems	MEDIUM	05/88
61	SRV Discharge Line Break Inside the Wetwell Airspace of BWR Mark I & Mark II Containments	MEDIUM	03/87
65	Component Cooling Water System	HIGH	07/86
66	Steam Generator Requirements	NEARLY RESOLVED	11/85
67	Steam Generator Staff Actions	MEDIUM	12/87
68	Loss of AFW Due to AFW Steam HELB	HIGH	10/87
70	PORV and Block Valve Reliability	MEDIUM	02/87
77	Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains	HIGH	08/86
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM	09/86
82.	Beyond Design Bases Accidents in Spent Fuel Pools	MEDIUM	09/86
83	Control Room Habitability (CRH)	NEARLY RESOLVED	TBD*
84	CE PORVs	NEARLY RESOLVED	TBD*
86	Long Range Plan for Dealing With Stress Corrosion Cracking in BWR Piping	NEARLY RESOLVED	04/86
91	Main Crankshaft Failure In Transamerica DeLaval, Inc. emergency diesel generators	NEARLY RESOLVED	TBD*

*Schedule To Be Determined

Table 7. Generic Issues Scheduled for Resolution

(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
93	Steam Binding of Auxiliary Feedwater Pumps	HIGH	06/87
94	Additional Low-Temperature-Overpressure Protection For Light Water Reactors	HIGH	10/86
99	RCS/RHR Suction Line Interlocks on PWRs	HIGH	TBD
101	BWR Water Level Redundancy	HIGH	09/86
103	Design For Probable Maximum Precipitation	NEARLY RESOLVED	07/86
105	Interfacing System LOCA at BWRs	HIGH	07/87
A-29	Nuclear Power Plant Design for the Reduction of Vulnerabilityto Industrial Sabotage	MEDIUM	05/87
A-30	Adequacy of Safety Related d.c. Power Supplies	HIGH	02/86
B-5	Ductility of Two Way Slabs and Shells and Buckling Behavior of Steel Containments	MEDIUM	09/87
B-6	Loads, Load Combinations, Stress Limits	HIGH	10/86
B-17	Criteria For Safety Related Operator Actions	Being resolved by HF-01, Issue HFPP-4.3	
B-55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	12/85
B-56	Diesel Reliability	HIGH	05/86
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	11/87
B-64	Decommission Nuclear Reactors	NEARLY RESOLVED	TBD
C-8	Reactor Main Steam Line Isolation Valve Leakage Control Systems	HIGH	12/86
I.A.2.6(1)	Long Term Upgrading of Training and Qualifications—Revise Regulatory Guide 1.8	Being resolved by HF-01, Issue HFPP 1.2	
I.A.4.2(1)	Research on Training Simulators	HIGH	10/85
I.A.4.2(4)	Review Simulators for Conformance	Being resolved by HF-01, issue HFPP-3.3	
I.B.1.1	Organization and Management of Long Term Improvements	—	—

Table 7. Generic Issues Scheduled for Resolution

(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
I.B.1.1(1)	Prepare Draft Criteria	Being resolved by HF-01, issues HFPP-6.1 and 6.3	
I.B.1.1(2)	Prepare Commission Paper	Being resolved by HF-01, issues HFPP-6.1 and 6.3	
I.B.1.1(3)	Issue Requirements for Upgrading Management and Technical Resources	Being resolved by HF-01, issues HFPP-6.1 and 6.3	
I.B.1.1(4)	Review Responses to Determine Acceptability	Being resolved by HF-01, issues HFPP-6.1 and 6.3	
I.C.9	Long Term Program Plan for Upgrading of Procedures	Being resolved by HF-01, issues HFPP-4.2 and 4.4 and HF-02	
I.D.3	Safety System Status Monitoring	MEDIUM	12/87
I.D.4	Control Room Design Standard	Being resolved by HF-01, issue HFPP-5.3	
I.D.5(5)	Disturbance Analysis Systems	Being resolved by HF-01, issue HFPP-5.4	
II.E.4.3	(Containment) Integrity Check	HIGH	10/86
II.E.6.1	Test Adequacy Stud	MEDIUM	05/88
III.D.3.1	Radiation Protection Plans	HIGH	11/85
B. Non-NRR Issues			
3	Set Point Drift in Instrumentation	NEARLY RESOLVED	01/86
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	NEARLY RESOLVED	07/86
119	Piping Review Committee Recommendations	NEARLY RESOLVED	TBD
I.A.3.3	Requirement for Operation Fitness	HIGH	12/85
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	MEDIUM	07/86
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	MEDIUM	07/86
I.D.5(3)	On-Line Reactor Surveillance System	NEARLY RESOLVED	09/88
I.F.1	Expand QA List	HIGH	12/85+
II.B.5(1)	Behavior of Severly Damaged Fuel	HIGH	12/87
II.B.5(2)	Behavior of Core Melt	HIGH	12/87

Table 7. Generic Issues Scheduled for Resolution

(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	MEDIUM	07/87
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	MEDIUM	TBD
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	12/86
II.J.4.1	Revised Deficiency Report	NEARLY RESOLVED	08/86
C. Human Factors Program Plan (HFPP) Issues			
HFKPP Subtask Numbers			
1.1	Policy Statement on Engineering Expertise on Shift and Evaluate Effectiveness of Policy Statement	HIGH	TBD
1.2	Revise and Evaluate Change to Regulatory Guide 1.8	HIGH	TBD
1.3	Develop a Means to Evaluate Acceptability of NPP Personnel Qualifications Program	HIGH	TBD
1.4	Review and Evaluate Industry Programs	HIGH	TBD
2.1	Evaluate Industry Training	HIGH	TBD
2.2	Evaluate INPO Accreditation Program	HIGH	TBD
2.3	Revise Standard Review Plan Section 13.2.3	HIGH	TBD
3.1	Develop Job Knowledge Catalogue	HIGH	07/86
3.2	Develop Licensing Examinations Handbook	HIGH	TBD
3.3	Develop Criteria for NPP Simulators	HIGH	03/87
3.4	Training Requirements Package (Revise 10 CFR 55 and RGs 1.149 and 1.8)	HIGH	TBD
3.5	Develop Computerized Exam System	HIGH	TBD
4.1	Inspection Module for upgrading procedures	HIGH	TBD
4 2	EOP Effectiveness Evaluation	HIGH	02/8i6
4.3	Criteria for Safety-Related Operator Actions (GI #B-17)	HIGH	TBD

+Schedules may extend beyond date shown.

Table 7. Generic Issues Scheduled for Resolution

(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
4.4	Guideline for Upgrading Other Procedures	HIGH	12/85
4.5	Applications of Artificail Intelligence	HIGH	TBD
5.1	Local Control Stations	HIGH	TBD
5.2	Annunciators	HIGH	TBD
5.3	Evaluate Operational Aid Systems	HIGH	TBD
5.4	Computers and Computers Displays	HIGH	TBD
6.1	Development of Regulatory Position on Management and Organization	HIGH	04/86+
6.2	Evaluate Criteria for SALP Reviews	HIGH	TBD
6.3	Revise Standard Review Plan Section 13.1	HIGH	11/86
6.3	Revise Standard Review Plan Section 13.1	HIGH	11/86
D. NRR Maintenance and Surveillance Program Plan (MSPP) Issues			
MSPP Subtask Numbers (Phase I)			
I.1	Survey Current Maintenance Practices	HIGH	TBD
I.2	Maintenance Performance Indicators	HIGH	TBD
I.3	Monitor Industry Activities	HIGH	TBD
I.4	Participate in Standards Group	HIGH	TBD
I.5	Maintenance and Surveillance Program Integration	HIGH	TBD
I.6	Analysis of Japanese/U.S. NPP Maintenance Programs	HIGH	TBD
I.7	Maintenance Personnel Qualifications	HIGH	TBD
I.8	Human Factors in In-Service Inspection	HIGH	TBD
I.9	Human Error in Events Involving Wrong Unit Wrong Train (GI #101)	HIGH	TBD
MSPP Subtask Numbers (Phase II)			
II.0	Phase II Tasks To Be Determined After Resolution of Phase I	HIGH	TBD

Human Factors

Staffing and Qualifications

The staff completed technical resolution of a number of generic safety issues affecting nuclear power plant staffing and personnel qualifications. (See Table 6.) The results of this work were published in NUREG/CR-4051, "Assessment of Job-Related Educational Qualifications for Nuclear Power Plant Operators," and NUREG/CR-4248, "Recommendations for NRC Policy on Shift Scheduling and Overtime at Nuclear Power Plants." The staff also completed TMI Action Plan Item I.A.3.4, dealing with the feasibility of licensing additional operations personnel. No necessity for additional personnel licensing was identified. In addition, the staff published NUREG/CR-3739, "The Operator Feedback Workshop: A Technique for Obtaining Feedback from Operations Personnel," and NUREG/CR-4139, "The Mailed Survey: A Technique for Obtaining Feedback from Operations Personnel," which will affect plant staffing, personnel qualifications and training.

The staff prepared a Final Commission Policy Statement regarding the need for engineering expertise on shift which allows licensees to combine the functions of the senior reactor operator and the shift technical advisor, thus permitting the integration of engineering expertise into the customary operating crew. Following Commission approval on September 12, 1985 (see 50 Fed. Reg. 43621, October 1985), the staff initiated implementation of the policy by reviewing relevant Technical Specifications for two facilities.

Revised Regulatory Guide 1.8 endorsing the minimum qualification standards established by ANSI/ANS 3.1-1981 for licensed operators was published for public comment during the report period. The Guide also allows for a determination of qualifications using a performance-based training model in lieu of the prescriptive criteria in ANSI/ANS 3.1. The staff is revising the Regulatory Guide to incorporate public comment.

On March 20, 1985, the Commission issued a Policy Statement on Training and Qualifications of Nuclear Power Plant Personnel (50 FR 11148), endorsing the Training Accreditation Program managed by the Institute of Nuclear Power Operations (INPO). The INPO program encompasses the elements of performance-based training and may be adequate to ensure that personnel have qualifications commensurate with the performance requirements of their jobs. The Commission has withheld action on promulgating training requirements during a period of evaluation. The staff is evaluating the results of the INPO accreditation program to determine whether the industry's voluntary efforts will assure qualifications that meet or exceed the minimum standards of Regulatory Guide 1.8.

Training

The Policy Statement described in the previous section is expected to ensure that plant operating personnel receive proper and adequate training to do their jobs.

The staff continued to visit utilities whose training programs are under review by an INPO accreditation team. The visits provide the staff with a better understanding of the industry's accreditation process. The staff has developed review criteria and procedures that will ensure that utility training programs include the five critical elements called for by the Policy Statement.

With the issuance of the Policy Statement on Training and Qualifications, NUREG/CR-4285, "An Approach to Team Skills Training of Nuclear Power Plant Control Room Skills," and NUREG/CR-4344, "Instructor Skills Evaluation in Nuclear Industry Training," the staff completed technical resolution of TMI Action Plan Items I.A.2.2, "Training and Qualifications of Operations Personnel," I.A.2.3, "Administration of Training Programs," I.A.2.5, "Plant Drills," and I.A.2.7, "Accreditation of Training Institutions."

The staff participated in the hearings on the adequacy of the operator training program in connection with the restart of Three Mile Island Unit 1 (Pa.) and also prepared testimony on S.16, the "Moynihan Bill", concerning the establishment of a National Academy for training nuclear power plant personnel. Requalification training programs for 27 facilities were also reviewed. Members of the staff prepared and delivered papers on training and qualification issues at a number of conferences and symposia during the report period.

Operator Licensing

Reactor operator licensing examinations are scheduled and administered through the NRC Regional Offices. During fiscal year 1985, NRC issued 401 new licenses and 535 license renewals for reactor operators, and 484 new licenses and 873 renewals for senior reactor operators. Regional Office personnel also conducted requalification examinations at 16 facilities and granted 85 instructor certifications.

The staff continued audits and program reviews in accordance with established guidelines and standards to assure regional examination consistency. "Knowledges and Abilities Catalog for Nuclear Power Plant Operators: Pressurized Water Reactors," NUREG-1122, was issued to aid examiners in constructing job-related examinations. A BWR generic catalog is expected to be developed in fiscal year 1986. The staff formulated and issued for public comment a rulemaking package containing proposed revisions to 10 CFR 55 and three Regulatory Guides (Regulatory Guide 1.8—Qualification and Training of Personnel for Nuclear Power Plants, Regulatory Guide 1.149—Nuclear Power Plant Simulation Facilities for use in Operator License Examinations, and Regulatory Guide 1.134—Medical Evaluation of Licensed Personnel for Nuclear Power Plants). Final Commission action on this package is expected during fiscal year 1986. The computerized Examination Question Bank was made fully compatible with IBM-PC microcomputer capability and the data base was made available to the public. In addition, NRC placed a contract for establishing criteria and procedures to test and evaluate simulationfacilities for examining operators and senior operators. This contract is scheduled for completion in fiscal year 1987.

Procedures

The NRC long term program for upgrading emergency operating procedures (EOPs) is in the implementation stage. Shortly after the TMI accident, the staff embarked on an effort to improve the technical accuracy and completeness of the EOPs and to incorporate human factors principles into the presentation of the technical material. Owners groups for all four vendors of nuclear power plants have satisfactorily reanalyzed accidents and transients and developed generic technical guidelines for their plants. In coordination with this effort, NRC issued a long term plan requiring all plants to revise EOPs based on approved technical guidelines and NRC guidance on incorporating effective human factors practices into procedure design.

The industry-wide program for the implementation of revised EOPs has made significant progress during the fiscal year. The required Procedures Generation Packages for almost all of the plants have been submitted to the NRC. These packages describe the applicant/licensee's programs for adapting the generic technical guidelines to the individual plant and the plans to use human factors considerations in the development of procedures. The NRC program allows implementation of upgraded procedures prior to completion of NRC review, so all plants are expected to be using upgraded emergency operating procedures within the next year.

The staff is auditing the implementation of EOPs at selected plants to evaluate the effectiveness of the program. During the report period, audits were performed at the Palo Verde Nuclear Generating Station (Ariz.) and the Millstone Nuclear Power Station (Conn.). Several more audits will be performed before an assessment of the NRC procedures program is made.

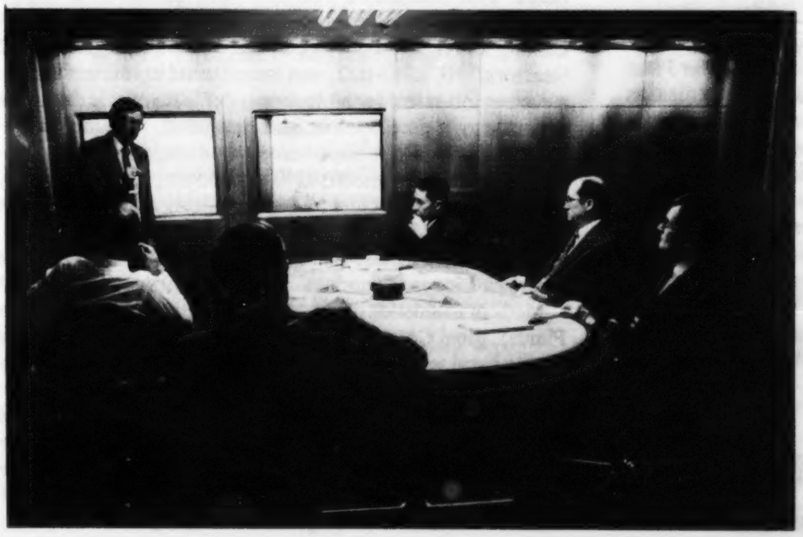
The staff has also prepared a revised inspection module as part of the EOP upgrade program. This will provide guidance to NRC resident inspectors for evaluating the implementation of upgraded EOPs. The revised module is expected to be implemented next year.

The original review of the generic technical guidelines identified certain unresolved technical issues. The staff has continued working with the owners groups to encourage further improvements in accident recovery strategies. To date, each of the four vendor owners groups has submitted revised technical guidelines; these are in various phases of review. These technical guidelines are also an important source of information for the selection of parameters for the Safety Parameter Display System to be incorporated into nuclear power plant control rooms.

Besides developing improvements in EOPs, the staff is studying the need to improve other operating and maintenance procedures.

Man-Machine Interface

The staff continues to evaluate the human factors aspects of man-machine interfaces to minimize design-induced errors in nuclear power plants. Generic Letter 82-33 (Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability") sets forth the basic requirements for detailed control room design reviews (DCRDRs) and for the Safety Parameter Display System (SPDS). The staff received 10 plans, representing 13 operating units, for detailed control room design reviews. By the end of fiscal year 1985, various utilities had initiated 61 detailed control room design reviews, representing 119 units. The staff conducted 26 in-progress audits, reviewed and commented on the 10 plans mentioned above, received 20 summary reports (25 units), issued 20 safety evaluation reports (26 units), conducted 16 pre-implementation audits (21 units), and completed control room preliminary design analyses for three applicants for operating licenses. In addition, the staff received 10 SPDS safety analysis reports (11 units), issued 21 safety evaluation reports (34 units), and conducted SPDS audits for 17 units. DCRDR and SPDS reviews will continue through fiscal year 1987.



NRC review of upgraded emergency operating procedures continued in 1985, facilitated by the new emergency Operations Center in NRC Headquarters in Bethesda, Md. In the photo, NRC Chariman Nunzio J. Palladino receives a briefing from Bernard H. Weiss of the NRC emergency response staff. In the new facility, plant and public safety information can be called up on two video monitors (on wall in background). The panel at left holds maps of all nuclear power plant sites and nearby populated areas for instant display as required.

As part of its continuing evaluation of human factors in the man-machine interface during 1985, the NRC conducted detailed control-room design reviews for 13 operating units, issued safety evaluation reports on 26 units and conducted 26 in-progress audits, among other measures. Here, Dan Jones of the Division of Human Factors Technology, Office of Nuclear Reactor Regulation, joins a plant representative during a procedures audit of an operating plant.

The staff issued safety evaluation reports for 45 units regarding "Data and Information Handling Capability" based on the generic implications of the Salem ATWS events (Generic Letter 83-28). In addition, the staff provided recommendations for improving the human engineering of the man-machine interfaces in the Technical Support Center and Emergency Operations Facility at Donald C. Cook Nuclear Power Plant, Units 1 and 2, Arkansas Nuclear One, Unit 1, Catawba Nuclear Station, Units 1 and 2, McGuire Nuclear Station, Units 1 and 2, and Washington Nuclear Power System, Unit 2.

The Man-Machine Interface Element of NUREG-0985, "NRC Human Factors Program Plan," was revised to reflect a growing concern about: (1) the nature of regulatory guidance appropriate for addressing the use of advanced technologies in upgraded systems, and (2) the necessity or desirability of imposing new human factors requirements on existing plant designs. The need for regulatory guidance and action will be determined for local control stations (outside main control room), improved and advanced annunciator systems, advanced control rooms and the use of computers. As the nuclear industry has been developing initiatives in these areas, the staff will coordinate its efforts with theirs and evaluate their programs.

Management and Organization

The NRC continued to evaluate management and organization at nuclear power plants. In its continuing effort to develop objective measures of nuclear utility performance, the staff published NUREG/CR-3737, "An Initial Empirical Analysis of Nuclear Power Plant Organization and its Effect on Safety Performance," reporting on the possible use of "safety indicators" as an evaluation tool. Publication of NUREG/CR-4125, "Guidelines and Workbook for Assessment of Organization and Administration of Utilities Seeking an Operating License for Nuclear Power Plants" completed the staff review of materials used by other organizations to evaluate management and organization; it provides a set of guidelines and a workbook to assist the staff in reviewing management and organization requirements (see Chapter 13 of the Standard Review Plan, NUREG-0800.)

The staff performed approximately 110 plant-specific actions to change technical specifications related to organization, management, staffing and training and 90 reviews of licensee plans for conducting post-trip reviews.

Maintenance and Surveillance

In response to Commission policy and planning guidance the staff developed a Maintenance and Surveillance Program Plan (MSPP). The overall purpose of the MSPP is to coordinate NRC and industry programs to evaluate maintenance effectiveness in the nuclear power industry. The Executive Director for Operations approved Phase I of the plan for implementation in January 1985.

A major accomplishment in the area of maintenance was the establishment of a cooperative working relationship between the staff and the Nuclear Utility Management and Human Resources Committee (NUMARC) and the Institute of Nuclear Power Operations (INPO). The staff reviewed and provided constructive comments on 10 NUMARC-generated maintenance performance indicators to monitor and track industry's self-initiated improvements.

With the introduction of the MSPP and the overall interest in maintenance activities, the Advisory Committee on Reactor Safeguards and various industry standards groups formed subcommittees on maintenance. NRC representatives of the MSPP staff became active participants on IEEE Working Group 3.3, "Maintenance Good Practices" and on the

ANSI/ASME NQA Working Group on "Maintenance, Repairing and Inservice Inspection."

The staff completed a joint study with the Japanese Ministry of International Trade and Industry to examine the requirements and experiences of maintenance programs in the U.S. and Japan. The study identified significant differences between the two countries' philosophy, practice, management and organization regarding preventive maintenance, regulation, and those socio-economic factors that may affect performance.

The loss of feedwater event that occurred at Davis-Besse (Ohio) in June 1985 was traced, in part, to inadequate maintenance practices. (See discussion under "Incident Response," in Chapter 8.) As part of a broader NRC study discussed elsewhere in this section, the staff performed a detailed survey of the Davis-Besse maintenance program and practices, and verified many of the weaknesses identified by the licensee. The study concluded that the utility remedial program was addressing all of the identified weaknesses observed by the team but that it was too early to assess their program's effectiveness.

Work on the following MSPP projects was initiated during fiscal year 1985:

- A study of human error in events involving wrong unit/wrong train.
- A study of the human factors aspects of ultrasonic/inservice inspection processes and a human factors evaluation of the inspectors' equipment.
- Development of a maintenance survey protocol to gather detailed information pertaining to specific utility maintenance programs and practices.
- Development of a computerized maintenance data base from which maintenance information on each nuclear power plant can be extracted.
- Development of NRC-generated maintenance performance indicators based upon publicly available information.

Unresolved Safety Issues

Section 210 of the Energy Reorganization Act of 1974, as amended, requires that the annual report of the Commission to the President and the Congress include progress reports on those items previously identified as "Unresolved Safety Issues" (USIs). A total of 27 USIs have been identified, and a final technical resolution has been achieved for 15 of these (see Table 8). Technical resolution of the remaining 12 USIs involves (1) development of technical findings and the incorporation of such findings into new licensing requirements in the NRC Regulations, Standard Review Plan, Regulatory Guides, or other official guidance; (2) provision for a plan for implementation of the technical resolution to plants in operation or under construction, if required; (3) preparation of a regulatory analysis of any new requirements and a review by the Committee to Review Generic Requirements (CRGR); (4) provision of a public comment period after CRGR review, followed by discussion and disposition of the comments received in a final report; and (5) provision for a second review of the resolution by the CRGR after public comments have been addressed.

The USIs that are being actively worked on are listed in Table 9, together with the present schedule for technical resolution. A summary of the status of USIs is published quarterly in NUREG-0606.

The following are progress reports on each of the Unresolved Safety Issues under active consideration during fiscal year 1985. For background on these issues, see the 1984 NRC Annual Report, pp. 26-31. No USIs were resolved during fiscal year 1985.

SUMMARY OF STATUS

PWR Steam-Generator Tube Integrity

Steam generator tube degradation in pressurized water reactors continues to be a matter of concern. (See the section on steam generators later in this chapter). The proposed resolution for this problem was discussed with the Commission in September 1984. Generic Letter 85-02 was issued to PWR licensees on April 17, 1985, requesting that the industry describe their overall program for assuring steam generator tube integrity, in order to allow the staff to compare their actions with staff recommendations. Also, the staff's program report, NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," was issued for public comment. The responses to these items are currently under review by the staff.

Systems Interactions

Adverse Systems Interactions are events that may jeopardize the independent functioning of nuclear plant systems. The staff and its contractor directed their efforts during fiscal year 1985 toward completing the technical work previously initiated to investigate the potential safety significance of these types of events and to explore possible ways of anticipating or uncovering these interactions.

The staff, aided by Oak Ridge National Laboratory, completed its investigation of operating experience at U.S. nuclear power plants as it relates to adverse systems interactions and issued an initial report, NUREG/CR-3922, "Survey and Evaluation of System Interaction Events and Sources," dated January 1985. A followup report, "Assessment of Systems Interaction Experience in Nuclear Power Plants" will be published soon. In the near future, the staff intends to issue a report entitled, "Review and Evaluation of Spatial Systems Interaction Studies," summarizing previous system interaction studies at certain utilities.

Two other reports on investigations of systems interaction search methods were issued in 1985. Brookhaven National Laboratory completed NUREG/CR-4207, "Fault Tree Application to the Study of Systems Interactions at Indian Point 3," dated April 1985. and Lawrence Livermore National Laboratory completed NUREG/CR-4179, "Digraph Matrix Analysis for Systems Interactions at Indian Point Unit 3," dated April 1985.

These studies complete technical work on the systems interactions issue. The proposed resolution of this Unresolved Safety Issue is under internal review.

Seismic Design Criteria

Rapid advancements in state-of-the-art technology in seismic design over the past decade have made it necessary to update the NRC acceptance criteria for seismic design of structures, systems, and components of nuclear plants. Lawrence Livermore Laboratory compared NRC Seismic Design Criteria with the current state-of-the-art knowledge and published their results in NUREG/CR1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria" May 1980. The staff proposes to change Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3 of the Standard Review Plan based on these recommendations. Incorporation of the proposed changes is expected to eliminate potential sources of non-conservatism and excessive conservatism, and result in seismic design criteria that reflect an up-to-date understanding of this technology. The two technical areas under consideration for changes to licensing criteria are soil-structure interaction analysis and design of free-standing, aboveground tanks. Soil-structure interaction analysis procedures are being revised to reflect recent advances in technology.

Containment Emergency Sump Performance

Following a loss-of-coolant accident (LOCA), long term heat removal is maintained by operation of residual heat removal (RHR) pumps and containment spray pumps. These pumps draw water from the containment emergency sumps in PWRs, and from RHR intakes located in BWR suppression pools, or wetwells. Safety concerns related to post-LOCA operation have been investigated through extensive full scale experiments, plant surveys and analyses. The staff's technical findings, along with information received during the public comment period, are reported in NUREG-0897, Revision 1, "Containment Emergency Sump Performance," and have been used to revise Regulatory Guide 1.82 and Standard Review Plan Section 6.2.2.

The value/impact of implementing these revised guidelines has been evaluated through regulatory analysis and public comments have been received. The results are reported in NUREG-0869, Revision 1, "USI A-43 Regulatory Analysis." The staff concluded that a requirement for industry-wide backfit to operating plants or plants under construction is not supported by the regulatory analysis and that revised regulatory guidelines should be required for new construction permit applications only. Since changing of thermal insulation on primary coolant system piping and components is an ongoing plant activity, the staff is planning to issue a generic information letter to all licensees highlighting the safety significance associated with a change to fibrous insulation materials. Issuance of these final documents will complete the resolution of USI A-43.

Station Blackout

The loss of all alternating current (a.c.) electric power (from both offsite and on-site sources) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require a.c. power supplies and on the ability to restore a.c. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event that could result in unacceptable consequences (e.g., severe core damage).

The staff's proposed resolution for USI A-44, which includes a proposed rulemaking and a new regulatory guide, was reviewed by the Committee to Review Generic Requirements (CRGR) during 1984. Subsequent to the CRGR review, the staff modified its technical position, based on an extensive review of updated information on losses of off-site power derived in cooperation with the Electric Power Research Institute. In addition, the staff issued a summary technical report, NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants" for comment.

The staff also received a proposal from the Nuclear Utility Group—an organization representing most of the utilities with nuclear power plants—on Station Blackout (NUGSBO). The NUGSBO position is that rulemaking on station blackout is not required because "station blackout does not represent a significant risk to public health and safety," but that improved protection would be provided by voluntary industry initiatives.

The Commission considered the staff proposal for resolution of USI A-44 at an initial meeting on September 11, 1985, but had not completed its deliberations at the close of the report period.

Shutdown Decay Heat Removal Requirements

The staff is continuing to evaluate the adequacy of systems for safely removing decay heat from a reactor core during shutdown and to assess the value and the impact of alternative measures for improving the reliability of those systems. Numerous tasks and subtasks are needed to accomplish these objectives—including system reliability assessments, system engineering feasibility studies, thermal-hydraulic analyses, power plant characterizations, reviews of emergency operating procedures, and evaluation of the vulnerability of the systems to special emergencies such as fire, flood, earthquake and sabotage.

Table 8. Formerly Unresolved Safety Issues for Which A Final Technical Resolution has been Achieved

Title		*Report Number*	*Date*	*Implementation Status*
A-1	Water Hammer	NUREG-0927 Rev. 1 NUREG-0933 Rev. 1	March 1984	No new requirements for operating plants. Revised SRP sections address requirements for any new application (see NUREG-0993), Rev. 1).
A-2	Asymmetric Blowdown Loads	NUREG-0609	November 1980	Resolution on final six operating plants will be based on leak before break in lieu of meeting criteria in NUREG-0609.
A-6	Mark I Short Term Program	NUREG-0408	December 1977	Complete
A-7	Mark I Long Term Program	NUREG-0661 NUREG-0661 Suppl. 1	July 1980	Licensees have designed and are installing modifications to meet the Commission's Order date for each operating plant with Mark I containment. Modifications have been completed on more than three-fourths of the 22 plants affected.
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808	August 1981	Implemented as part of the OL review of each Mark II containment
A-9	Anticipated Transients Without Scram	NUREG-0460 Volume 4	September 1980	The final rule was published in the Federal Register (49FR5752) on June 26, 1984. Guidance for implement ation on all plants is included in the final rule
A-10	BWR Feedwater Nozzle	NUREG-0619	November 1980	Complete
A-11	Reactor Vessel Material	NUREG-0744, Rev. 1	October 1982	Implementation on a case-by-case basis as needed
A-12	Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1	September 1982	No implementation on operating plants required
A-24	Qualification of Class 1E Safety Related Equipment	NUREG-0588 Rev. 1	July 1981	Implementation in accordance with the new rule 10CFR 50.49 is continuing. Any exemptions to completion by November 30, 1985 arebeing reviewed by the Commission
A-26	Reactor Vessel Pressure Transient Protection	NUREG-0224	September 1978	Complete
A-31	Residual Heat Removal	SRP 5.4.7	1978	Implemented as part of the review for each operating license application. No backfit to operating reactors is planned

Title		*Report Number*	*Date*	*Implementation Status*
A-36	Control of Heavy Loads Near Spent Fuel	NUREG-0612	July 1980	Detailed implementation for each licensee is continuing
A-39	SRV Dynamic Loads	NUREG-0802	September 1982	Implemented as part of the OL review of each Mark II and Mark III containment
A-42	Pipe Cracks in Boiling Water Reactors	NUREG-0313 Rev. 1	July 1980	Actions for each licensee on a case-by-case basis in accordance with operating experience

Work has been completed on developing questions for the qualitative screening of light water reactor decay heat removal capabilities and preliminary assessment of decay heat removal (DHR) capability of operating and soon-to-be operating light water reactors. In the latter case, screening led to the selection of seven plants for further analysis of their shutdown decay heat removal capabilities. An analysis using up-to-date probabilistic risk assessment methodology that considers operating transients and accidents as initiating events and also considers special emergency events such as fire, flood, earthquake and sabotage has been completed for two plants. Similar studies are being performed on the remaining five plants. The involved licensees assisted in obtaining all information necessary to model their plants. Each study will identify plant-specific vulnerabilities regarding decay heat removal needs resulting from transients, accidents, and special emergencies. Conceptual designs are being developed and costs estimated for various alternatives to reduce risk in each plant; the value/impact of such alternatives will be determined.

The staff expects to complete its work on the seven plant studies during the first half of fiscal year 1986 and prepare a summary report. This information will form the basis for the staff's proposed generic technical resolution of USI A-45.

Seismic Qualification of Equipment In Operating Plants

The design criteria and methods employed for the seismic qualification of mechanical and electrical equipment in nuclear power plants have changed significantly during the history of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and to perform intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to assure the ability to bring the plant to a safe shutdown condition following a seismic event. This issue (USI A-46) entails investigation of alternative procedures for assuring seismic adequacy of equipment, in lieu of requiring qualification to current licensing requirements, as defined in Standard Review Plan Section 3.10, Regulatory Guide 1.100 and IEEE Standard 344/1975.

The staff evaluated the various methods available for verifying seismic adequacy of equipment in operating nuclear power plants. The use of seismic experience data, as proposed by the Seismic Qualification Utility Group (SQUG), proved to be the most reliable and cost-effective way of verifying seismic adequacy. A Senior Seismic Review and Advisory Panel (SSRAP) independently established, with some caveats and exclusions, the feasibility of using experience data to verify the seismic adequacy of operating plant equipment. This review was based on data concerning eight categories of equipment collected by the SQUG.

The staff concluded from its USI A-46 investigation that there are two principal areas of concern: the adequacy of equipment anchorages and supports, and the functional capability of electrical relays. The staff developed a proposed resolution and implementation procedure for USI A-46 which provides general guidelines for walk-through inspection of equipment anchorages and review of functional capability of electrical relays. Technical findings are documented in draft NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants." In July 1985, the Committee to Review Generic Requirements (CRGR) recommended that the Regulatory Analysis for Proposed Resolution of USI A-46 and draft NUREG-1030 be issued for public comment. These documents were made available to the public in September 1985.

The provision for a generic implementation program for those utilities that choose to participate in the Generic Group is a key element of the proposed resolution. The SQUG is committed to developing and coordinating such a program and is continuing to develop detailed implementation procedures for use by participating utilities. Following the general guidelines provided in the NRC-proposed resolution, the SQUG is continuing the following activities:

- Develop equipment anchorage guidelines (joint program with EPRI).
- Develop electrical relay review guidelines.

Table 9. Schedule for Resolution of Current Unresolved Safety Issues

Task No.	*Unresolved Safety Issue*	*Scheduling for Issuing Staff Report "For Comment" (as of Sept. 30, 1985)*	*Schedule for Issuing Final Staff Report (as of Sept. 30, 1985)*
A-3,4,5	PWR Steam Generator Tube Integrity	Completed April 1985 1985	Not scheduled
A-17	Systems Interactions	November 1985	July 1986
A-40	Seismic Design Criteria	Not scheduled	Not scheduled
A-43	Containment Emergency Sump	Completed May 1983	December 1985
A-44	Station Blackout	December 1985	December 1986
A-45	Shutdown Decay Heat Removal Requirements	November 1986	November 1987
A-46	Seismic Qualification of Equipment in Operating Plants	Completed Sept. 1985	June 1986
A-47	Safety Implications of Control Systems	March 1986	October 1986
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns	—	Not scheduled
A-49	Pressurized Thermal Shock		
	Rulemaking	Completed Feb. 1984	Completed July 1985
	Complete Technical Resolution	November '85	March 1986

- Collect qualification test data (joint program with EPRI and NRC/RES).
- Develop plant walk-through procedures.
- Conduct trial walk-throughs.
- Develop format and data sheets for recording and reporting walk-through results.
- Conduct workshops for participating utilities.

The generic implementation program will provide a common set of procedures, record-keeping and guidelines which will be implemented by each plant separately after adequate training and familiarization in workshops. SQUG and SSRAP will be responsible for assuring that the generic procedures and guidelines address the issue in a comprehensive way and that the individual utilities are provided with the necessary understanding to complete the reviews of specific plants in an acceptable manner. In accordance with requirements of the proposed resolution, the SSRAP will be continued as an independent review group throughout the implementation process. Both the SSRAP and the NRC staff have been working closely with the SQUG and will continue to monitor the ongoing SQUG activities and will audit implementation on selected plants in the Generic Group.

The generic implementation method is preferred by the staff for the implementation of USI A-46, but provisions for implementation of USI A-46 for individual utilities not participating in the Generic Group are also included in the proposed resolution. For these plants, the staff will review both the inspection report and the final report and will audit all plant-specific reviews prior to the final NRC approval.

Safety Implications of Control Systems

The staff is performing systematic evaluations of control systems that typically are used during normal startup, shutdown and on-line power operations of nuclear power plants from each of the four nuclear steam supply system vendors (i.e., Babcock and Wilcox, Westinghouse Corp., Combustion Engineering, and General Electric Co.). The purpose of this study (USI A-47) is to identify control systems whose failure could either cause transients or accidents to be potentially more severe than those identified and analyzed in the licensee's final Safety Analysis Report, adversely affect any assumed or anticipated operator action during the course of an event, cause technical specification limits to be exceeded, or cause transients or accidents to occur at a frequency in excess of those established for abnormal operational transients and design-basis accidents. The Idaho National Engineering Laboratory has evaluated PWR and BWR designs and issued final reports. In addition, the Oak Ridge National Laboratory is completing its evaluation of the Babcock & Wilcox and Combustion Engineering pressurized water reactor designs and expects to issue final reports in October 1985.

These studies have identified several control system failures that could cause transients leading to steam generator or reactor vessel overfill, overcooling, overpressure or overheating events. The staff is analyzing the safety significance of the control system failures that have been identified.

The analyses of the BWR and the two PWR plants is complete and final reports are under review by the NRC staff. The analysis for the fourth plant design (also a PWR) is in the early stages of review. A proposed staff resolution which includes recommendations for operating plants and for future plants is under development.

Hydrogen Control Measures and Effects Of Hydrogen Burns on Safety Equipment

Large quantities of hydrogen may be generated and released to the containment following a degraded core accident. The ignition of this combustible gas could threaten the integrity of the containment and/or its equipment (USI A-48).

Several technical programs were initiated in 1980 to investigate the control of large amounts of hydrogen in the containment. As a result, NRC has promulgated rules concerning Mark I, II and III containments for boiling water reactors and ice-condenser containments for pressurized water reactors. A final rule was published on December 2, 1981, requiring that Mark I and II containments be filled with nitrogen during reactor operation. The final rule for Mark III and ice-condenser containments was published on January 25, 1985, requiring systems that can control an amount of hydrogen equivalent to that generated by a reaction of 75 percent of the fuel cladding with water.

The staff will complete technical resolution of USI A-48 by reviewing the hydrogen control systems for Sequoyah (ice condenser) and Grand Gulf (Mark III) and publishing an NRC generic summary report on hydrogen control. To support these remaining activities, the NRC and industry have sponsored extensive research programs. Large scale hydrogen combustion tests were completed at the Nevada Test Site (NTS) in early 1984. The staff has evaluated the NTS premixed combustion tests and expects to complete evaluation of the NTS continuous hydrogen injection tests by early 1986. In addition, the Mark III Owners Group is sponsoring a one-quarter scale hydrogen test program. Results of the scoping tests for this program are under evaluation and the entire test matrix was scheduled to be completed by the end of calendar year 1985.

Pressurized Thermal Shock

Pressurized Thermal Shock (PTS) events involve unintended rapid cooling of the steel reactor pressure vessel to a low temperature concurrent with or followed by repressurization of the water inside the vessel. If a flaw or crack exists at a location where the toughness of the vessel's inner surface has been decreased excessively by the neutron irradiation that occurs during normal power generation, severe PTS events could cause rupture of the vessel with potential melting of the enclosed nuclear core.

After extensive analyses performed by the NRC staff and several nuclear industry groups, the staff concluded that: (1) the risk due to PTS events is related directly to the "reference temperature," which is a measure of ductility loss and is determined from the reactor vessel material properties, the high energy neutron irradiation at the reactor vessel wall, and the duration of reactor operation; and (2) the risk due to PTS events is acceptably low if the "reference temperature" has not exceeded a certain specified screening limit, which has been defined by the staff.

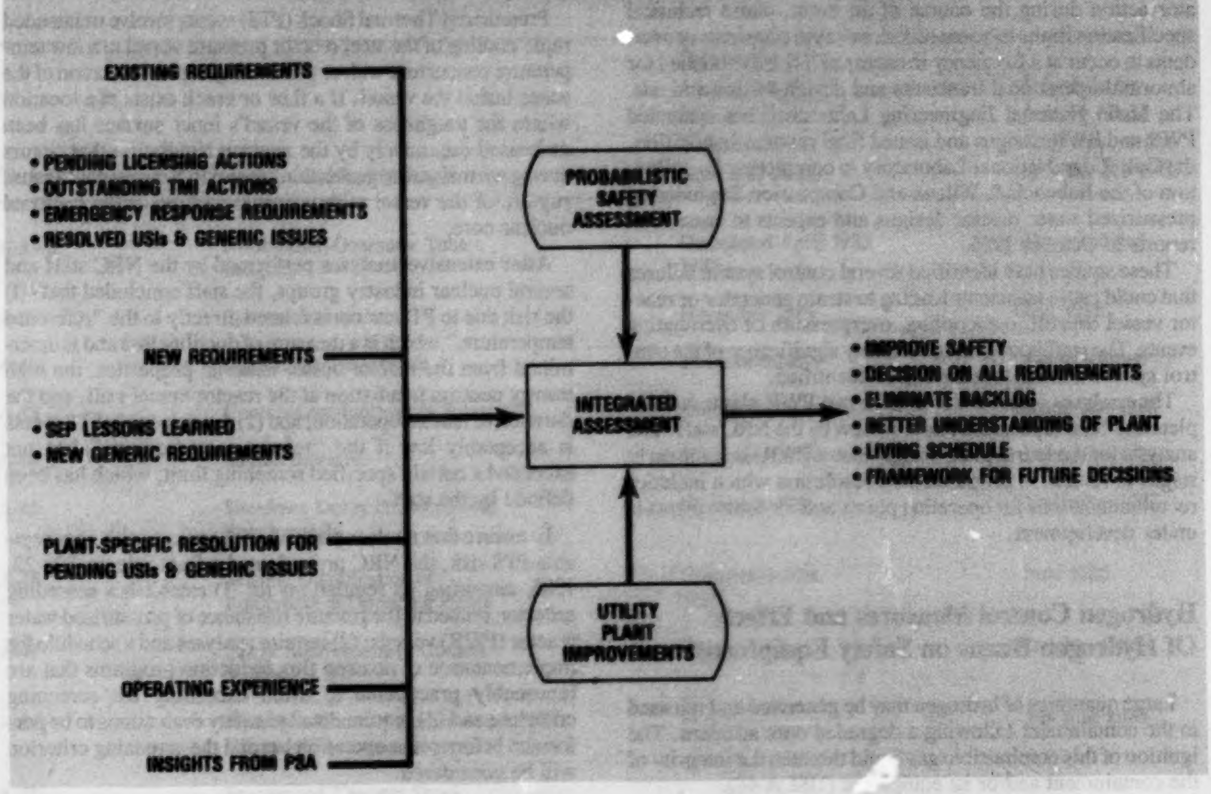
To ensure that nuclear plants do not operate with unacceptable PTS risk, the NRC promulgated a final rule on July 23, 1985, amending its regulations to: (1) establish a screening criterion related to the fracture resistance of pressurized water reactor (PWR) vessels; (2) require analyses and a schedule for implementation of neutron flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion; and (3) require detailed safety evaluations to be performed before plant operation beyond the screening criterion will be considered.

The staff also has completed prototype analyses for three nuclear plants to develop the bases for guidance to licensees who will be required to perform plant-specific PTS risk analyses to justify any proposed operation beyond the screening limit. This guidance, which also indicates the acceptance criteria that the staff will use in reviewing acceptability of operation beyond the screening limit, will be published for public comment in late 1985 or early 1986, well before any plant will near the screening limit.

Safety Reviews

TMI Action Plan

The accident at Three Mile Island Unit 2 (Pa.) in 1979 led to a thorough review of NRC regulatory and licensing requirements for nuclear power plants. A TMI Action Plan was issued, as NUREG-0660, and the requirements approved for implementation at plants in operation or under construction were later clarified, in NUREG-0737. TMI Action Plan requirements for plants under construction are being implemented as



INTEGRATED SAFETY ASSESSMENT PROGRAM

EXISTING REQUIREMENTS
• PENDING LICENSING ACTIONS
• OUTSTANDING TMI ACTIONS
• EMERGENCY RESPONSE REQUIREMENTS
• RESOLVED USIs & GENERIC ISSUES
NEW REQUIREMENTS
• SEP LESSONS LEARNED
• NEW GENERIC REQUIREMENTS
PLANT-SPECIFIC RESOLUTION FOR PENDING USIs & GENERIC ISSUES
OPERATING EXPERIENCE
INSIGHTS FROM PSA
PROBABILISTIC SAFETY ASSESSMENT
INTEGRATED ASSESSMENT
UTILITY PLANT IMPROVEMENTS
• IMPROVE SAFETY
• DECISION ON ALL REQUIREMENTS
• ELIMINATE BACKLOG
• BETTER UNDERSTANDING OF PLANT
• LIVING SCHEDULE
• FRAMEWORK FOR FUTURE DECISIONS

part of the licensing process, while those for operating reactors are confirmed by NRC orders. Items not covered by NUREG-0737 have been addressed in NUREG-0933, which sets priorities for generic issues.

Supplement 1 to NUREG-0737, which delineates the requirements for emergency response capabilities, was sent to all licensees on December 17, 1982. Implementation schedules were discussed with the utilities at regional meetings, and by June 12, 1984, the schedules were confirmed by issuance of Confirmatory Orders for all licensed light water reactors.

At the end of fiscal year 1985, approximately 7,000 TMI-related items had been reviewed by NRR. Approximately 1,500 open items associated with the TMI Action Plan were left to be reviewed, about half of which will be the subject of future regional verification inspections. Another thousand or so items whose reviews are complete are also scheduled for regional verification inspections. About 1,800 items remain to be verified. Of the 1,500 open items, approximately 30 percent are involved with plants that have been licensed since the TMI accident, and 20 percent involve changes to Technical Specifications. Approximately 20 percent of the Technical Specification changes are for plants that have been licensed since the TMI accident. On the average, there are about 16 open TMI items per plant, with most of the items being associated with Supplement 1 to NUREG-0737.

Integrated Safety Assessment Program

In a policy statement published in the Federal Register on November 15, 1984, the Commission described a trial program to evaluate all licensing issues on a given operating reactor to establish effective and efficient implementation schedules for any necessary plant modifications. This Integrated Safety Assessment Program (ISAP—see figure) replaces the Systematic Evaluation Program (SEP) and the National Reliability Evaluation Program (NREP). (See *1984 NRC Annual Report*, pp. 31, 32.)

The pilot program was implemented in early 1985 for two plants—Millstone Unit 1 and Haddam Neck, both in Connecticut. The staff conducted a detailed screening review to identify the "topics" to be evaluated in ISAP. That review considered all of the pending licensing actions, SEP experience, unresolved safety issues, and licensees' plant improvements. The results of the screening review were given to Northeast Utilities in a letter dated July 31, 1985. The review identified 80 "topics" to be evaluated in ISAP for Millstone Unit 1 and 70 for Haddam Neck, as well as plant modifications and ongoing engineering projects that would be conducted independently of ISAP.

Following the screening review, the licensee began to submit safety analyses for each of the Millstone Unit 1 topics. Millstone Unit 1 will be the first plant to conduct an integrated assessment in the ISAP. (Figure.) In addition, the licensee initiated separate evaluations of plant operating experience and the plant-specific probabilistic safety analyses, to identify other factors to be considered in the integrated assessment. In July 1985, the licensee submitted a probabilistic safety analysis report for Millstone Unit 1 that identifies the dominant contributors to risk for that facility. A similar study is under way for Haddam Neck, and the results of that analysis will be reported in early 1986.

The ISAP evaluations for Millstone Unit 1 and Haddam Neck are scheduled to be completed by the ends of calendar year 1985 and 1986, respectively. When the Millstone Unit 1 evaluation is complete, the staff will forward the results to the Commission, along with a recommendation concerning the extension of ISAP to other operating reactors.

Probabilistic Risk Assessment

Probabilistic risk assessments (PRAs) continue to provide valuable insights into the importance of certain potential safety issues and a mechanism for the systematic identification of strengths and weaknesses in nuclear power plant designs and operation. During fiscal year 1985, the staff completed technical evaluations of GESSAR, Zion (Ill.), Millstone-3 (Conn.), and Shoreham (N.Y.).

Based on the GESSAR-II PRA review and the Zion Probabilistic Safety Study review, the staff recommended measures for reducing the vulnerability and risk of core damage accidents. In both cases, the recommendations addressed the loss of electrical power (a.c.) from off-site sources needed for normal operation and safety systems, and the availability of emergency on-site a.c. and d.c. power.

The submittal and review of the PRA for GESSAR-II (standardized BWR design with a Mark III containment) was the first application of the Severe Accident Policy Statement (50 FR 32138, August 8, 1985) in the licensing process. The staff recommendations sent to General Electric specifying changes required for the approval of the GESSAR-II design, include:

- Adding an Ultimate Plant Protection System which is a backup system requiring no emergency electrical power.
- Increasing the capability in d.c. power sources to improve plant response to a loss of off-site power situation.
- Adding a Hydrogen Control System.

The staff completed the Zion Risk Evaluation and Insights report and performed a regulatory analysis of its recommendations based on the PRA review. These recommendations addressed two risk-important scenarios, loss of coolant accident outside containment (Interfacing Systems LOCA) and loss of all normal and emergency a.c. electrical power (Station Blackout). The recommendations sent to the licensee for comment were:

- The diesel-driven containment spray pump should be modified to remove its dependency on emergency a.c. power and thus increase assurance of containment cooling capability following loss of all a.c. power.
- The motor-operated valves in the Residual Heat Removal System should be tested to increase the assurance of proper valve position and the integrity of the high pressure/low pressure boundary outside of containment.

Limiting conditions of operation (LCO) are contained in Technical Specifications for each plant. These specifications dictate, among other things, the frequency of testing components and systems and the allowed outage time for maintenance or repair while the plant is operating. Risk-based analyses and PRAs are being used to evaluate changes in the specifications on plant-specific and generic bases. Some of these activities relate to the Seabrook Station (N.H.), the Byron Generating Station (Ill.), and the BWR Owners Group proposal to modify the Reactor Protection System Technical Specifications. The Byron review is nearly complete and a draft report by Brookhaven National Laboratory will be completed in fiscal year 1986. The ongoing risk-based evaluation of selected Seabrook Technical Specifications is expected to be complete in early 1986. This is being done concurrently with the review of Standard Technical Specifications in the licensing process. The BWR Owners Group has proposed modifications, based on risk analyses, to the allowed outage time and surveillance test intervals contained in the Reactor Protection System Technical Specifications. An evaluation of the justification will be completed in January 1986. On the system level, Brookhaven National Laboratory in support of the staff, evaluated the effect of changes in the allowed outage time for the Auxiliary Feedwater System in PWRs using reliability and risk assessment techniques.

In addition, the staff:

- Reviewed the industry-sponsored Oconee (S.C.) PRA. During the course of the PRA, the utility voluntarily made modifications to deal with the possibility of flooding inside the plant induced by external flooding.
- Reviewed the Yankee Rowe (Mass.) PRA. It will be one of the first industry-sponsored PRAs being reviewed primarily in-house by NRR staff with a greatly decreased role of outside contractors.

In generic PRA activities, the NRC staff issued two draft reports for comment: NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide," and the companion document NUREG/CR-3485, "PRA Review Manual." These reports include consideration of both internal events (failures of reactor systems) and external events (e.g., seismic events, floods, high winds). These documents will be completed in early fiscal year 1986. Insights into PRA results have been provided to a wide audience through NUREG/CR-3852, "Insights into PRA Methodologies," and an annually updated report, "Probabilistic Risk Assessments Insights" (Draft NUREG/CR-4405) expected to be published in the first quarter of fiscal year 1986.

Probabilistic assessments are used routinely in setting priorities for new issues according to their safety significance, and for allocating resources. Similarly, probabilistic assessments are used in weighing alternative solutions to generic safety issues. In addition, the probabilistic assessments provide an important basis, along with deterministic engineering judgment, for the regulatory analyses of new requirements proposed by the staff.

Severe Accident Policy

Severe nuclear accidents are those in which substantial damage is done to the reactor core, whether or not there are serious off-site consequences. On September 19, 1984, the staff forwarded to the Commission for review and approval a recommended "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants." The Commission approved the Policy Statement on July 30, 1985 and published it in the *Federal Register* on August 8, 1985 (50 FR 32138).

On the same date, the staff issued the final draft of the companion report, "NRC Policy On Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulations" (NUREG-1070). This report provides the rationale and supporting data for the severe accident policy statement. It includes a discussion of numerous inter-related ongoing severe accident programs. Among them are: the Severe Accident Research Program (NUREG-0900); the Source Term Program (NUREG-0956); the development of Safety Goals and the Probabilistic Risk Assessment Reference Document (NUREG-1050); the resolution of Unresolved Safety Issues and Generic Safety Issues; and the integration of insights from IDCOR, foreign reactor and regulatory experience as well as the staff review of new reactor designs. A separate chapter of NUREG-1070 provides an overview of, and staff response to, public comments and the views and recommendations received from the ACRS. The report also includes a short appendix on the treatment of uncertainty in the severe accident program and a more detailed appendix on current information bearing on the need for generic design changes or further regulatory changes affecting nuclear power plants. The latter provides a rationale for the differential policy treatment of existing and future plants and an up-to-date information base to support a number of critical premises or assumptions underlying the basic strategies of the Policy Statement.

The Policy Statement, with its emphasis on procedures and criteria for staff review of new standard plant designs, is intended to facilitate and stabilize review procedures (See "Standardization," earlier in this chapter.). A staff review of severe accidents for the GESSAR II design for forward referenceability was published in July 1985 (NUREG-0979). (See discussion below.) The staff has also been working with the Preliminary Design Approval (PDA) application for Westinghouse Electric Corporation's advanced pressurized water reactor design—RESAR-SP/90.

Risk From Seismic Events

The risk of severe nuclear reactor accidents due to seismic events has been calculated in recent probabilistic risk assessments (PRAs) for Zion (Ill.), Indian Point (N.Y.), Limerick (Penn.), Millstone 3 (Conn.), and the General Electric standard plant, GESSAR II. In these PRAs, the calculations of core-melt frequency and consequences to the public are based on estimates of the probability of equipment failure at varying levels of earthquake severity (fragility distributions), and the likelihood that earthquakes of these varying levels will occur in the vicinity of the plant (hazard function). By combining the hazard function with the fragility distributions for the relevant plant safety features, the PRA analyst can estimate the core-melt frequency, identify the plant safety systems that are major contributors to seismic risk, and determine the range of earthquake magnitude that is most likely to cause a core melt accident. In fiscal year 1985, the staff completed detailed reviews of the seismic analyses for Millstone Unit 3 (NUREG-1152, to be published) and GESSAR II, (NUREG-0979, Supplement 3).

Based on reviews and extensive interaction with the ACRS regarding GESSAR II, the staff has gained important new insights about seismic risk. While the assumptions and methods used in these PRAs vary significantly, there are several common features of the results. First, seismic events are a significant, and in some cases dominant, contributor to overall risk, particularly the risk of early fatalities. However, the results generally are below the risk target levels of the proposed safety goals. Second, seismic risk is estimated to result from events significantly more severe than the Safe Shutdown Earthquake—the extreme, but rare, seismic event that the plant is designed to withstand. This result indicates that there is appreciable safety margin in the seismic design of existing nuclear power plants. Finally, the uncertainties in the methods and assumptions used in seismic risk calculations are large, and the numerical results should be used with caution. The staff's decision-making on issues related to seismic risk has been based in part on the PRA results, complemented by engineering judgment and prudence.

The issue of seismic vulnerabilities is an important focus of the forthcoming staff effort to implement the Commission's Severe Accident Policy Statement. This statement provides for an integrated systematic approach to an examination of all operating plants not yet having had such an examination, in order to identify unique plant features that contribute significantly to severe accident risk. The experience gained in the Millstone 3 and GESSAR II reviews, as well as the earlier work on Zion, Limerick and Indian Point, provide important insights in evaluating seismic risk and will serve as a basis for the planned systematic search for unique plant vulnerabilities to severe accident risk.

Operational Safety Assessment

Assessment of the significance of unanticipated events at operating reactors involves NRC Headquarters and Regional

Offices, which provide prompt reviews and technical support on issues and events of possible immediate safety concern. In addition, the NRC staff reviews such events against existing licensing analyses, evaluates plant and operator performance during events, identifies generic safety implications, reviews licensee analyses, and evaluates corrective actions prior to plant restart.

In fiscal year 1985, the staff initiated a formalized program for the assessment of major reactor incidents. The first incident investigation team assembled under this program investigated the complete-loss-of-main-andauxiliary feedwater event that occurred at the Davis-Besse facility (Ohio) on June 9, 1985. The results of this investigation were published in NUREG-1154.

Other examples of operating reactor events occurring in fiscal year 1985 are:

- Major engine failures of emergency diesel generators at North Anna Unit 2 (Va.) in January and March, 1985.
- Containment tendon failures at Farley Unit 2 (Ala.) in January 1985.
- Common-cause problems with steam generator pressure instrumentation at Maine Yankee in August 1985.
- Loss of Emergency Core Cooling System due to inadequate design change control at La Salle Unit 2 (Ill.) on June 10, 1985.

Implementation of the ATWS Rule

An anticipated-transient-without-scram (ATWS) is an expected operational transient (such as loss of feedwater, loss of condenser, or loss of off-site power to the reactor) which is accompanied by a failure of the reactor trip system (RTS), a part of the plant protection system, to shut down the reactor. A failure of the RTS could result from common cause failures of redundant and identical RTS components such as logic circuits or actuation devices (e.g., circuit breakers on pressurized water reactors). ATWS events are a cause of concern because under certain postulated conditions they could lead to extremely high reactor coolant system pressures, core damage, and release of radioactivity to the environment. The latest precursor to an ATWS event was a failure of the automatic portion of the RTS at the Salem 1 nuclear generating station (N.J.) on February 25, 1983.

The Commission has amended its regulations to require improvements in the design and operation of light-water-cooled nuclear power plants to reduce the likelihood of failure of the RTS to shut down the reactor (scram) following anticipated transients and to mitigate the consequences of ATWS events. The specific equipment requirements are contained in 10 CFR 50.62 (known as "the ATWS rule") and are summarized below:

The first investigation team dispatched under the formal reactor incident assessment program initiated in 1985 visited the Davis-Besse nuclear power plant following the June 9, 1985 incident. Shown here are NRC and Toledo Edison officials during a tour of the plant in October 1985.

- All pressurized water reactors (PWRs) must have equipment, independent and diverse from the existing RTS, to automatically initiate auxiliary (emergency) feedwater and trip the turbine under conditions indicative of an ATWS.
- All Combustion Engineering and Babcock & Wilcox PWRs must have a diverse scram system.
- All Boiling Water Reactors (BWRs) must have an alternate rod injection system.
- All BWRs must have a standby liquid control system (SLCS) with a minimum flow capacity of 86 gpm. SLCS initiation must be automatic for plants granted a construction permit after July 26, 1984.
- All BWRs must have equipment to automatically trip the reactor recirculation pumps under conditions indicative of an ATWS.

Implementation of the above requirements will provide diversity to those portions of existing RTS designs where only minimal diversity is currently provided, thus reducing the potential for common cause failures of the RTS and resulting in a reduction in risk from ATWS events. A diverse scram system is not required for Westinghouse PWRs because they are less sensitive to severe ATWS sequences, principally because of the larger pressure relief capacity built into the Westinghouse plant design. Additional information concerning the ATWS rule is provided in the *Federal Register* (Volume 49/Number 124/June 26, 1984/ Rules and Regulations) and in SECY-83-293 "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," dated July 19, 1983.

Each nuclear power plant licensee was required to develop and submit by October 14, 1985, a proposed schedule for implementation of the applicable ATWS equipment requirements. Since the equipment required by the ATWS rule does not have to be designed and installed as safety-related, the Commission has issued applicable quality assurance guidance in

Generic Letter 85-06 "Quality Assurance Guidance for ATWS Equipment that is Not Safety-Related," dated April 16, 1985. The schedules for implementation of the ATWS equipment requirements will be evaluated for consistency with the goal of integrating new requirements into other existing plant programs, considering the unique status of each plant and the relative safety importance of the improvements.

Evaluation of Control Room Habitability

Since 1980, the Advisory Committee on Reactor Safeguards (ACRS) has held several meetings with the NRC staff to discuss the subject of control room habitability. These meetings have occasioned a number of ACRS letters expressing specific concerns, to which the staff has responded. On August 15, 1983, a plan, jointly developed by the Offices of Nuclear Reactor Regulation (NRR) and Inspection and Enforcement (IE), that would address the latest ACRS concerns and recommendations was approved. This program plan was implemented by the Control Room Habitability Working Group during fiscal year 1984.

During the present report period, the staff published the results of an Argonne National Laboratory-West (ANL) survey of licensee control room habitability practices at three plants as NUREG/CR-4149. This report dealt with how the evolution of control room designs has led to an increase in the complexity of the habitability systems. It further discussed how this complexity can create system maintenance concerns, e.g., leakage being measured through flow control and isolation dampers of the control room engineered safety feature (ESF) filtration and adsorption unit and the control room heating, ventilating and air conditioning (HVAC) system. Other problems were identified with the plant technical specifications, inconsistencies between as-built systems and the systems described in the plant FSAR, and licensee procedures for demonstrating system control, function and operability through testing. The report recommendations included consideration of use of radioprotective drugs for the control room crew and consolidation of NRC criteria on control room habitability into one document along with the bases for such criteria. Control room surveys will be conducted on an additional 12 plants in conjunction with regional inspections.

Occupational Exposure Data and Dose Reduction Studies

The staff has been tabulating the annual occupational doses at light water reactors (LWRs) since 1969 (see figure). Although both pressurized water reactor (PWR) and boiling water reactor (BWR) annual dose averages have fluctuated over the years, the overall trend between the mid-1970s and 1980 was one of increasing annual dose averages. Since 1980, however, these dose averages have leveled off at between 700 and 800 person-rems per unit for LWRs. In 1984, the average dose per unit for LWRs was 708 person-rems, a 6 percent decrease from the 1983 average. The average doses per unit for BWRs and PWRs were 1,003 and 552 person-rems, respectively. This is the 11th consecutive year in which the average BWR doses per unit have exceeded the average PWR doses per unit. Maintenance jobs which were large contributors to BWR doses included replacement of recirculation system piping, inspection and repair of intergranular stress corrosion cracking, Mark I torus modifications, and reactor vessel component inservice inspection (ISI). Steam generator maintenance and repair are a major source of occupational exposure at PWRs.

The 1984 dose tabulation includes data from 27 BWRs and 51 PWRs. This total reflects the addition of 5 new plants (LaSalle 1 (Ill.), San Onofre Unit 2 (Cal.) , St. Lucie Unit 2 (Fla.), Summer Unit 1 (S.C.), and Susquehanna Unit 1 (Pa.) and the deletion of two older plants (Humboldt Bay (Cal.) and Indian Point Unit 1 (N.Y.)) which have been defueled, with no plans to operate them again.

The NRC has several ongoing contracts with Brookhaven National Laboratory (BNL) in the area of occupational dose reduction at LWRs. One recently initiated study evaluates the magnitude of plant contamination problems and determines the estimated annual collective dose and cost savings from minimizing contaminated plant areas. Other ongoing contract studies by BNL are determining the cost-effectiveness of dose reduction techniques, identifying and evaluating high-dose maintenance tasks at LWRs, comparing occupational doses at U.S. and foreign LWRs, and compiling a research data base on dose reduction projects at nuclear power plants.

Achieving ALARA in Occupational Radiation Exposure

NRC efforts towards developing effective measures to reduce radiation exposures to levels "as low as reasonably achievable" (ALARA) in the operation of commercial power reactors include: regulatory action, radiological safety reviews, radiation protection/ALARA inspections, and interaction with industry. All proposed regulatory actions—over the full range of regulations, guidelines, and generic and other safety issues—require that occupational doses incurred as a result of implementing these actions be considered along with other decision criteria. Thus, license applications and amendments require a staff radiological safety/ALARA review employing NRC standards. Each NRC Region conducts inspections in radiation protection/ALARA to identify possible eficiencies and desirable improvements at each facility. In addition, the NRC staff has participated in cooperative efforts with industry to achieve mutual goals in radiation protection/ALARA.

Pursuant to the "Coordination Plan for Radiological Protection Activities," the staff continues to monitor power reactor industry radiation protection programs incorporating ALARA concepts. Under this cooperative agreement with the Institute of Nuclear Power Operations (INPO), NRR staff members observed the INPO process for evaluating radiation protection at power reactors and reviewed data trends for a number of radiation protection performance criteria. Preliminary evaluations of radiation protection data trends from both NRC and INPO information sources indicate a leveling and/or slight diminution of average annual doses at power reactors. This trend analysis covers the five year period of effort by INPO and the industry, from 1980 to 1984. Proposed regulations requiring ALARA programs and promulgating a form and content regulatory guide describing such programs have been held in abeyance pending the final review of the industry level of success in achieving ALARA-integrated radiation protection programs.

Radioactive Effluents Summary and Analysis

The program for implementing Radiological Effluent Technical Specifications (RETS) in operating reactors continued during the reporting period. By the end of fiscal year 1984, about 8 percent of the operating nuclear reactors had received technical approval for their specifications. Many of the plants implemented their RETS during fiscal year 1984. The balance of those that have been approved are to be implemented during 1985. In addition, regulatory action on the remaining 20 percent is expected during 1985.

As of July 1985, Radiological Effluent Technical Specifications (RETS) had received technical approval at all operating nuclear reactors and were implemented at 80 percent of them. Essentially all operating reactors were expected to be operating under the RETS by early 1986.

Licensees submit periodic reports on radioactive effluents and radiological environmental monitoring as part of the RETS license requirements. Semiannual reports contain detailed summaries characterizing the radioactive gaseous and liquid effluents released from the plant to the environment and also quantify solid radioactive wastes shipped off the site. These reports include calculations of the radiation doses from these effluent releases to members of the public off-site. NRC annually collates these individual plant summaries in two publications: "Radioactive Materials Released from Nuclear Power Plants" and "Population Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites."

In addition to the semiannual effluent reports, licensees submit an annual radiological environmental operating report. This report contains the results of the extensive weekly and monthly monitoring programs required by the plant's RETS and records when, if ever, radioactive contamination above natural background is detected outside the plant boundaries. The semiannual effluent reports and the annual radiological environmental operating reports for all operating plants are available for public inspection in local Public Document Rooms.

Testing Methods for Activated Charcoal

Laboratory analysis of activated carbon in ventilation filter systems at nuclear plants is required by the Technical Specifications for Engineered Safety Features (ESF). Because of the safety importance of these filter systems, the frequency of such tests is at least once per refueling cycle (about 18 months), or more frequently, depending on the hours of operation of the system and the volume of chemical and physical processes that could degrade the performance of the activated carbon. In addition, plant procedures may require that many non-ESF activated carbon filter units installed in normal ventilation systems be tested. The purpose of the laboratory analysis is to ensure that the carbon adsorbers are capable of operating at an efficiency at least equal to that assumed in the NRC staff's Safety Evaluation Report (SER). If the laboratory analysis shows that the carbon material has a removal efficiency for radioiodine less than the value specified in the Technical Specifications or by plant-specific procedures, then the carbon in the filtration system must be replaced.

In 1982, the Committee on Nuclear Air and Gas Treatment of the American Society of Mechanical Engineers (ASME) invited a number of laboratories from the United States and elsewhere to participate in a round-robin testing program of several nuclear-grade activated carbon samples, both new and used. The disparity of results of the round-robin tests indicated significant disparities. (See 1984 NRC Annual Report, p. 40.) The evaluation of the laboratory test method for activated carbon continued in fiscal year 1985. Eleven laboratories from the U.S., Canada, South Korea, United Kingdom, and the Federal Republic of Germany participated in the second interlaboratory test comparing new and used carbon. In general, the test results showed excellent reproducibility or precision within the individual laboratories but poor agreement between the laboratories, with greater correlation between certain laboratories for the used carbon than the new. It was agreed at a workshop held in June 1985, that test results indicate a calibration problem with test instrumentation. Other problem areas identified in the test method during the workshop included accuracy of mass-flow meter devices and purity of the test reagents being used. As a result of the workshop, the testing laboratories intend to form a working group and work together to identify the principal problems of the laboratory test method.

The NRC's contractor, EG&G-Idaho, has focused its attention on the calibration problem, the completion of sensitivity tests on the various parameters of the American Society for Testing Materials test method, and on potential problems related to the purity of the source material. In September 1985, EG&G-Idaho, with the assistance of experts on the measurements of gas flow and relative humidity from the National Bureau of Standards, conducted a calibration workshop at its test facility in Idaho. Four commercial U.S. laboratories participated in the workshop. Gas flow and humidity measurement instrumentation was calibrated for each of the five participating laboratories. Work is continuing in the evaluation of the test method during fiscal year 1986 with additional testing of standard carbon samples.

Source Terms: Releases of Radionuclides in Severe Accidents

The methodology being developed by NRC contractors to evaluate severe accident releases ("source terms") was reviewed independently by a study group of the American Physical Society, and their findings were published in *Reviews of Modern Physics 57,* No. 3, July 1985. Improvements needed to satisfy the study group's recommendations have not yet been completed. The NRC staff, however, is closely following the work being done by the contractors and has begun developing plans for adopting the new methods into its review procedures, as those methods become available.

The first trial use of the new methodology will be in the assessment of the environmental impacts of severe accidents to be reported in the Draft Environmental Statement for the South Texas Project, Units 1 and 2. This assessment is being performed twice, once using the latest source terms available, and once using source terms adapted from the rebaselined Reactor Safety Study (RSS). Accident consequence calculations using the new source terms associated with South Texas predicted no early fatalities (radiation exposures resulting in death within one year), although early fatalities were computed using RSS source terms.

The staff and representatives of the Industry Degraded Core Rulemaking Program (IDCOR) participated in technical exchange meetings throughout 1985. These meetings have served to identify a list of significant differences between the methods of estimating risk from severe accidents used by IDCOR and by the NRC staff. When these issues are fully resolved, a methodology under development by IDCOR will be used to search for risk "outliers," as prescribed in the Commission's Severe Accident Policy statement.

GESSAR-II Design Improvement Study

Publication in July 1985 of the results of its assessment of potential design improvements for the General Electric standard plant, GESSAR-II (NUREG-0979, Supplement 4) concluded the staff's review. This study required two years of effort by numerous NRC technical groups and consultants. This is the first application of the NRC's policy statement on severe accidents (NUREG-1070) which demonstrates how severe accident policy applies to future standard plant designs. With this policy guidance, the staff reviewed design conformity with current requirements and in a context of concerns deriving from unresolved safety issues and generic issues; a full-scope probabilistic risk assessment was done. The staff also evaluated potential design improvements to assess whether cost-beneficial means could be found for reducing severe accident risk. General Electric was asked by the staff to evaluate an extensive list of over 70 potential design improvements. General Electric reported the results of its evaluation in NEDE-30640, June 1984, concluding that none of the potential design modifications was cost-beneficial. In addition, the staff and its consultants independently performed detailed studies searching for means—such as the use of filtered, vented containment systems—to mitigate the major risk contributors for the GESSAR-II design.

As a result of these studies, the NRC staff recommended that the GESSAR-II design should include the General Electric-proposed Ultimate Plant Protection System upgraded for seismic events, with some additional improvements— including hydrogen control with dedicated independent power and improved d.c. battery capability and charging. The Ultimate Plant Protection System is intended to provide diverse core and containment heat removal capability, a vital feature of defense-in-depth design philosophy for severe accident prevention. In reaching its conclusions, the staff used the results of the cost-benefit studies for screening purposes, while final decisions were based on engineering judgement focused on cost-effective risk and risk uncertainty reduction. The staff concluded that further design modifications beyond those recommended are not justified because of the low level of risk predicted for the GESSAR design. The risk level for off-site health effects predicted by the staff for GESSAR-II is orders of magnitude lower than that being considered in the Commission's proposed Safety Goal (48 *FR* 10772, March 14, 1983).

The staff plans to perform similar studies on the Westinghouse standard plant design, SP-90.

Alternative Disposal of Radioactive Wastes

NRC regulations permit any applicant for or holder of a license to apply to the Commission for approval of proposed procedures to dispose of licensed material in a manner not otherwise authorized in the regulations. Nuclear power plant licensees have been applying with increasing frequency for approval under 10 CFR 20.302 to dispose of slightly radioactive wastes by means other than burial at licensed low-level waste disposal facilities. The alternative disposal means chosen generally depends upon the nature of the materials; incineration is the usual choice for waste oils, and burial, either on-site or in a municipal landfill, for solid wastes. Materials to be disposed of have included waste oil, roofing materials, miscellaneous wood, feedwater heaters, sandblasting sand, secondary-side resins, sewage, wastewater treatment and settling basin sludges, deposits from fossil-fueled boiler fireboxes, and slightly contaminated soils. One disposal was made to a facility for chemically hazardous wastes. The predominant reactor-originated radionuclides in the wastes are usually Cs-137, Cs-134, Co-60, Co-58 and Mn-54. Concentrations have ranged from hundredths to tens of picocuries per gram. The staff reviews each request and documents its findings. In most cases, it is expected that no member of the public will receive more than one millirem per year from any disposal. The applications have been either for one-time disposal of a specified collection of materials, or for approval of procedures to dispose of limited quantities of a specified type of material on an annual basis. In fiscal year 1985, NRC staff approved 11 such applications from nuclear power plant licensees. Applicants have been advised that approval may also be necessary from State and/or local governments.

Radioactive Waste Incineration At Nuclear Power Plants

Compliance with 10 CFR 20.305 requires that licensees obtain NRC approval prior to incinerating licensed material. The use of volume reduction processes incorporating incineration, or both incineration and drying for certain types of radioactive wastes, has been approved for several nuclear power plant licensees. In incineration, wastes are burned to ashes, significantly reducing the volume to be processed, packaged, transported, and disposed of at a licensed land burial site. The drying processes use the hot exhaust gases from the incinerator to remove moisture from noncombustible wastes to likewise reduce their volume.

Incineration of contaminated oil in the auxiliary boilers has been approved for the Brunswick plants (N.C.) and the FitzPatrick plant (N.Y.). Incineration of contaminated oil in the auxiliary boiler had been previously approved for the Oconee plants (S.C.). Incineration and drying of evaporator concentrates, dry active wastes and contaminated oil in a dedicated volume reduction system has been approved for the Byron plants (Ill.). Incineration and drying of evaporator concentrates, spent secondary resins, dry active wastes and contaminated oil in a dedicated volume reduction system has been proposed but not yet approved for the Oconee plants. Incineration of dry active wastes and contaminated oil using a vendor-supplied mobile system has been proposed but not yet approved for the Dresden plants (Ill.). Previously submitted applications for the use of incineration and drying in dedicated volume reduction systems at the Braidwood plants (Ill.) and at the Vogtle plants (Ga.) were under evaluation by the staff at the close of the report period.

Spent Fuel Pool Modifications

Various licensees have submitted a significant number of requests to the staff to increase on-site spent fuel storage capacity; the number of these requests is expected to increase. This increasing demand for space to store spent fuel assemblies is brought about by the unavailability of off-site storage facilities to licensees who are nearing exhaustion of their available on-site storage capacity. Facilities for reprocessing spent fuel are not expected to be operational in the near future, so that a reprocessing option is not currently viable for relief of the dwindling storage capacity.

Since most licensees used low density spent fuel storage racks in their original spent fuel pool (SFP) designs to prevent criticality in the spent fuel pool, the more recent practice using high-density racks would provide a considerable increase in storage capacity. The proposal to replace the low-density racks with high density racks involves a change in Technical Specifications with concomitant staff review and issuance of a Safety Evaluation Report (SER) and an Environmental Impact Appraisal (EIA).

In the staff's evaluations for requested increases in spent fuel pool storage capacity, the design-basis accident scenarios normally considered are the fuel-handling accidents and, where applicable, the cask tip/drop, gate drop, heavy load drop, and tornado missile accidents. Hypothetical accident scenarios beyond the design basis—such as possible stored fuel oxidation propagation following a large pool water loss, initiated by external events or refueling cavity seal failure—are being addressed as part of Generic Safety Issue 82.

To assure a complete review of the licensee's Safety Analysis Report (SAR) for modification of the SFP storage configuration, and to speed up the review process, the staff developed a Standard Review Plan (SRP) for review of the occupational radiation protection program of the SFP modification plan. The SRP provides uniform technical guidance to licensees and staff. Some of the details addressed in the SRP derive from issues raised by the Atomic Safety and Licensing Board and by intervenors during relevant hearings; they also reflect the experience gained by the staff during 42 facility design modification reviews. The staff is considering SRP modifications for other review areas.

Transfer and Storage of Spent Fuel

On July 13, 1985, the Virginia Electric and Power Company (the licensee) applied for an amendment to revise the operating licenses for the North Anna Power Station, Units 1 and 2, to permit the receipt and storage of 500 spent fuel assemblies from the Surry Power Stations, Units 1 and 2. The Atomic Safety and Licensing Board (ASLB) held an evidentiary hearing on May 21-22, 1985 in Charlottesville, Va., and issued an Initial Decision, LBP-85-34, on September 3, 1985, authorizing the Director of NRR to approve the licensee's application.

The ASLB concluded in its decision that, contrary to affirmations of the intervenor, Concerned Citizens of Louisa County (CCLC), an environmental impact statement need not be prepared because: (1) the staff's Environmental Assessment, both as a matter of law and as supplemented by the Board's findings, adequately evaluated the probability and consequences of shipping accidents, (2) CCLC failed to challenge the Safety Evaluation Report's analysis of sabotage, and (3) the record established the low probability of either a sabotage attack's being undertaken or being successful, and that, even if such an attack were successful, the impact upon the public health and safety and upon the environment would be very small. Further, the ASLB concluded that, contrary to the assertion of CCLC, the trans-shipment proposal does not involve unresolved conflicts concerning alternative uses of available resources and that no basis exists for concluding that the dry cask storage alternative is preferable to the trans-shipment proposal.

"Feed-and-Bleed" Capabilities At Davis-Besse Unit 1

On June 9, 1985, a temporary loss of all feedwater event occurred at the Davis-Besse Unit 1 (Ohio), a Babcock and Wilcox 177-FA PWR, raised-loop facility. While auxiliary feedwater was recovered by the operators in a timely manner, the "feed-and-bleed" capability of the plant came into question. The significance of this is that if feedwater is not restored relatively soon, and if other emergency means for decay heat removal (such as feed-and-bleed procedures) are not available, then a core meltdown could eventually occur. The causative factors in this event were unique to the Davis-Besse plant, because the relatively low shutoff head of the High Pressure Injection pumps combined with the single power-operated relief valve (PORV) on the pressurizer to compromise its feed-and-bleed capability.

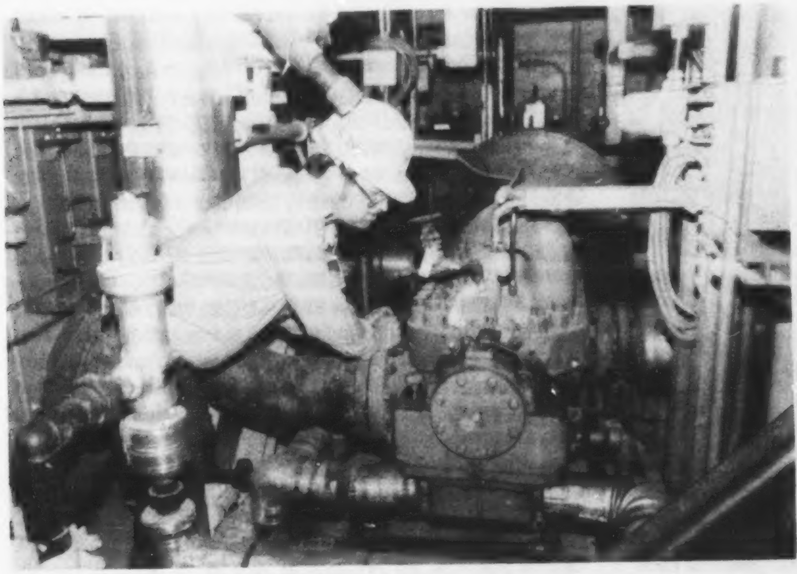
NRR contractors at the Los Alamos National Laboratory assessed the feed-and-bleed capability of Davis-Besse in a hypothetical situation where auxiliary feedwater would not have been restored. The results of these studies (LA-UR-85-3083, "Rapid Response Analyses of the Davis-Besse Loss-of-Feedwater Event on June 9, 1985) demonstrated that, had auxiliary feedwater not been restored, the operators would have had at least an additional 15 minutes (or 34 minutes from the start of the event) to start feed-and-bleed procedures which would prevent core uncovery. NRR has performed numerous studies in-house concerning feed-and-bleed at Davis-Besse with the RELAP5 computer program and the Nuclear Plant Analyzer. These studies, which will be documented in an internal NRR memorandum, included variations in the number of make-up pumps used for feed-and-bleed, the addition of a second PORV, full-power studies and the effect of having the PORV or the safety relief valves cycling prior to the start of feed-and-bleed. In addition, simplified mass and energy balance calculations were performed to assess alternate modes of decay heat removal or core cooling. The NRR studies concluded that the addition of a second PORV would enhance the feed-and-bleed capability at Davis-Besse and diminish the reliance on the make-up system, as only one make-up pump would be required to prevent core uncovery. The results also indicated that feed-and-bleed would be successful if initiated within 20 minutes of a total loss-of-feedwater event at full rated power. The licensee and Babcock and Wilcox are currently reevaluating the feed-and-bleed initiation criteria and performing analyses to support any recommended changes to the plant operating procedures.

The staff is studying the generic significance of this event. Issues being evaluated include plant management, plant maintenance and the need for an additional PORV. The work in progress under Unresolved Safety Issue A-45, "Decay Heat Removal," is also relevant to the Davis-Besse event. The staff has described the analyses performed for Davis-Besse to the ACRS Subcommittee for Emergency Core Cooling Systems on two occasions.

Environmental Radioactive Contamination from Rancho Seco

In May 1984, the Sacramento Municipal Utility District reported that calculated doses to an off-site individual resulting from releases of radioactive liquid effluent from its Rancho Seco Nuclear Power Plant (Cal.) exceeded the exposure standards of 40 CFR 190 (as referenced in 10 CFR 20.105) for the period 1980 through the first quarter of 1984. This standard requires that a licensee provide reasonable assurance that doses to the whole body and organs, other than the thyroid, will not exceed 25 millirems in a year, and that the dose to the thyroid will not exceed 75 millirems during this same time period. The District failed to detect this situation because of an error in the computer code used to perform the dose calculations over this time period. The excessive doses resulted from leaks in the steam generators at the plant. The licensee has taken actions to bring releases within their authorized limits.

The NRC contracted the Oak Ridge National Laboratory to conduct an evaluation of the environmental contamination



NRC resident inspector Don Kosloff is shown examining one of the auxiliary pumps which failed during the June 9 incident at Davis-Besse (Ohio). The plant's two main feedwater pumps tripped while the plant was operating at 90 percent power. The reactor and the turbine tripped 30 seconds later and both main steam isolation valves closed, cutting off steam to the second main pump.

around the plant. The main objectives of this project were to identify the types and measure the concentrations of radionuclides in the environment. Laboratory scientists visited the plant site during November and December, 1984, to sample fish, frogs, game birds, beef, soil, silt, vegetation, and water for radioactivity; external radiation levels also were measured to assist in determining the extent of ground contamination.

The study team found elevated levels of cesium-134, cesium-137, and smaller amounts of cobalt-60 in the water and silt samples immediately below the area where the plant discharges its water into a nearby stream. These levels decrease as distance from the plant increases. Higher-than-background levels of radioactivity were also detected in samples of fish, beef, game birds, and vegetation.

Human consumption of fish and frogs is estimated to be the largest potential contributor to individual doses. Doses from other foods and from direct radiation from the stream and contaminated land adjacent to the creek contribute only a small amount to the total dose. Based on a land use census performed by the utility, one individual may have received a whole body dose of about 50 millirems. This dose is equivalent to half of one year's exposure to natural background radiation. The investigation is continuing and may lead to a change in these estimates.

Site Population Distributions

Site-related regulatory considerations—such as compliance with siting criteria, accident source evaluations, and evacuation assessments—require staff cognizance of population distributions and population projections in the vicinity of nuclear power plants. To maintain authoritative and accurate information, the staff has developed a complete set of population data for the 79 present and projected nuclear power plant sites in the United States. The data are based on the 1980 U.S. Census, and include population estimates out to 500 miles from any given site. Each site-specific population distribution has been coupled with a related growth rate data set developed for NRC by the U.S. Department of Commerce, Bureau of Economic Analysis. These growth rates may be used to project site populations up to the year 2010. The staff has developed a set of computer programs to facilitate editing and maintaining the population data base, and determining projections and specialized distributions.

Geosciences

Because of continuing uncertainties and new information concerning the structural geology and seismology of the central California coastal region, the staff has conditioned the Diablo Canyon nuclear plant's operating license with the requirement that the utility conduct additional geological, seismological, ground motion, and probability studies over the next three years to revalidate the seismic design. The staff, assisted by the U.S. Geological Survey (USGS), the Lawrence Livermore National Laboratory (LLNL), Brookhaven National Laboratory and the University of Nevada—Reno, will reevaluate this new information while it reviews the utility's work. The staff and its geological advisors recently conducted two geological reconnaissances of the Diablo Canyon site and region. Following conditional acceptance of its program plan, the utility has begun reevaluation studies.

Eastern Seaboard Seismicity

Most nuclear power plants are in the eastern or central United States, which, unlike California, is geologically an intraplate region, where relatively little is known about the sources and causes of earthquakes. This lack of knowledge has resulted in some controversy in the consideration of earthquake possibilities for purposes of nuclear power plant design. (See *1984 NRC Annual Report,*, p. 41.) The staff, aided by LLNL, has undertaken a program to characterize seismic hazard for this region on a probabilistic basis. The methodology uses expert opinion as the source of all the seismicity and ground motion data. LLNL recently published a report (LLNL Report UCID-20421) presenting probabilistic estimates of peak ground acceleration and response spectra at 10 test sites. These results are being compared with results obtained by other methods, including those developed by the USGS and the Electric Power Research Institute.

In the deterministic program, areas of relatively higher seismicity along the eastern seaboard are being studied to determine whether tectonic features and processes responsible for the seismicity can be identified and correlated. Field evidence and radiocarbon analyses suggest that in the past 3,000 to 3,700 years at least two earthquakes large enough to cause liquefaction preceeded the 1886 Charleston (S.C.) event. Based on the available evidence, the maximum recurrence interval for such earthquakes in the Charleston region is estimated to be about 1,500 to 1,800 years. Additional study of these features is expected to improve our assessment of the frequency of large earthquakes in the region. Further investigations should determine how far from Charleston liquefaction features can be found, and thus whether we can expect large earthquakes elsewhere in the coastal plain.

Other Seismic Regions

Because of the presence of many nuclear power plant sites, the NRC has been funding research in the Southern Appalachians, notably the Southern Appalachian Regional Seismic Network (SARSN) in eastern Tennessee and western North Carolina. Analysis of the data from the first 2 years of operation of this network indicates that, contrary to the historical record, the Valley and Ridge Province of eastern Tennessee is more active seismically than the Blue Ridge and Inner Piedmont. As in the Giles County Zone, most of the activity is occurring below the discontinuity on north-south faults under a NE-SW compressive regime.

Recent field investigations of the Meers fault in southwestern Oklahoma, approximately 11 miles northwest of Lawton, reveal what may be the only known example of recent major tectonic surface rupture in the central and eastern United States. The northwesterly-trending fault is relatively young, at least 26 kilometers in length and has up to five meters of apparent topographic offset. Although the fault has been known for a long time, it was previously assumed to be of Paleozoic age, like many of the faults in the midcontinent area. The region around the Meers fault has a low seismic activity, with no known events definitely associated with the fault. Seismographs placed near the fault have not detected any seismicity; however, it is not unusual for certain faults to be intermittently active. Applying standard formulas to available data indicates a possibility of generating a magnitude 6 to 7.5 earthquake. The implications of such an event on the Meers fault was assessed for existing nuclear power plant sites in the region. Ground motions from such an event were found to be less than the design basis earthquakes for these plants. The NRC is funding additional field work—including aerial photography, trenching, and radiometric age dating—in an attempt to establish the capability of the Meers fault. Two trenches across the fault were completed and geologically mapped this year. Evidence from the trenches—along with aerial photography, a study of geomorphology, and the radiocarbon dating of young soils which showed a cross-cutting relationship with the fault—suggests that the fault last moved between 500 and 2,000 years ago. If it is demonstrated to be capable (as defined in Appendix A of 10 CFR 100), the staff will then evaluate the relevance of that fact to an understanding of eastern U.S. seismicity.

Several nuclear power plant sites are located in the Pacific Northwest. Geological and geophysical evidence in this region indicates that the Juan de Fuca Plate is actively subducting under the North American Plate, although no direct evidence has been found of a large thrust-type of earthquake which is historically typical of most active subduction zones around the Pacific Ocean. Subduction in this region may be taking place aseismically, but, on the other hand, recent evidence gathered during a USGS research program, partially funded by the NRC, indicates that the Juan de Fuca Subduction Zone has many of the characteristics of subduction zones around the Pacific Ocean which have experienced major thrust earthquakes. The USGS will continue to study this problem as well as the potential volcanic hazard in the Pacific Northwest.

During the year, staff members visited Egypt, Ecuador and New Brunswick, Canada, to assist in geological and seismic aspects of nuclear plant regulation.

Foundations

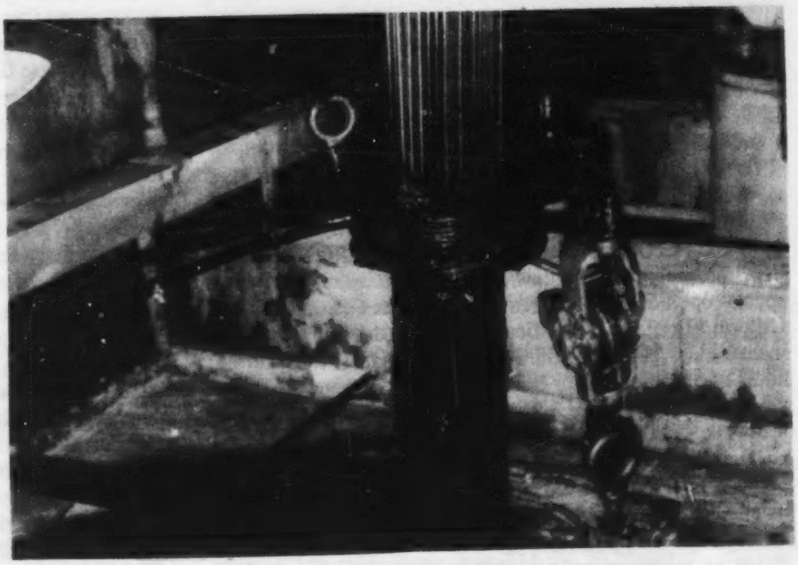
Vogtle Electric Generating Plant. Subsurface investigations at the Vogtle nuclear plant site (Ga.) have revealed the presence of a shelly limestone layer which has been subjected to extensive leaching and the formation of open solution cavities. The depth of the cavernous limestone layer below plant grade is approximately 95 feet. To provide stable foundations for all safety-related structures and piping, the applicant has completed extensive foundation excavation operations to remove the solutioned limestone layer and all soil materials above it, within the entire area where seismic Category I structures are located. Approximately 5 million cubic yards of soil and rock were removed and replaced with compacted granular soils consisting of clean medium-to-fine sands (SP) and sands with some silt (SP-SM). As backfilling proceeded, the foundations for the various safety-related structures and piping were constructed at the design-determined foundation levels until plant grade was reached. The verification activity—which included a confirmatory laboratory test program to establish maximum dry densities of the backfill and borings to investigate the already placed and compacted backfill—showed the compacted backfill soils to be very dense with very good foundation engineering properties.

San Onofre Nuclear Generating Station Unit 1. This plant, located in an area of significant earthquake potential on the Pacific coast near San Clemente, Cal., began commercial operation in 1968. Two larger units on the same site, San Onofre 2 and 3, began operation in 1984. As a result of the NRC's Systematic Evaluation Program, the licensee recognized a need to improve the seismic resistance of the earlier plant to levels comparable to those of the later plants. This upgrading required excavation adjacent to Unit 1 safety class structures so that additional buried foundations could be installed to resist design seismic forces. During the excavation through previously placed backfill soils, pockets and zones of under-compacted fill were discovered beneath some safety class facilities. The owners investigated the extent and degree of compaction of these soils to assess their support characteristics during and after a design seismic event. The concern is that the soils beneath these facilities will be permanently saturated during operation of the plant and might liquefy as a result of earthquake shaking.

As a result of the owners' study, a structural bridge was placed over the insufficiently compacted material to support safety equipment. The ends of the bridge derive their support from adjacent foundations or from the competent *in situ* San Mateo sand formation. In some instances, the backfill in place was found to be sufficiently compacted to resist liquefaction and settlement effects.

Structural Engineering

The failure of three field anchor heads for vertical prestressing tendons was discovered during a visual inspection of the Farley Unit 2 (Ala.) containment structure. The figure shows a field anchor head of the type which was found cracked and failed. Also shown is how the anchor head is attached to the steel wires for prestressing the concrete containment. In response to this discovery, the licensee established a comprehensive inspection program for Units 1 and 2, and the NRC staff formed a task group to assess this unanticipated problem. Both the licensee and the staff have concluded that the most likely cause of the failure appears to be hydrogen stress cracking, resulting from water entrapped on the anchor heads. The



During a visual inspection of the containment structure at the Farley Unit 2 nuclear power plant (Ala.), three "field anchor heads" for vertical prestressing tendons were found to have failed. The photos show a field anchor head of the type found cracked. The licensee established a comprehensive inspection program for Units 1 and 2, and the NRC staff has formed a task group to assess this unanticipated problem. The licensee and the NRC both concluded that the most likely cause of the failure was hydrogen stress cracking, resulting from water entrapped on the anchor heads.

licensee has removed, tested, and regreased 135 field anchor heads from Unit 2 and 130 from Unit 1. The NRC staff is preparing an evaluation report on the Farley problem and assessing whether there are any generic implications.

Prompted by the occurrence of the 1982 New Brunswick earthquake, the staff completed its seismic margin review of Millstone Unit 3 (Conn.), evaluating the plant's capability to withstand greater than design-basis earthquakes. The review relied upon the existing seismic PRA for Millstone Unit 3, and utilized hazard, fragility, and system information. This review confirmed the staff expectations that Millstone Unit 3 possesses considerable margin beyond the design basis earthquake.

A portion of the steel liner of the concrete containment structure at Millstone Unit 3 was damaged by fire during construction. This damaged portion has been cut out and subsequently repaired. The staff review of the applicant's engineering report and justification, consisting of two complex analyses of liner-stud interaction and sample tests of concrete in the affected area, confirmed the adequacy of the repaired structure.

BWR Thermal Hydraulic Stability

The staff has investigated the possibility of thermal-hydraulic instabilities inducing large oscillations in the neutron flux in a boiling-water-reactor (BWR) core, resulting in critical heat flux safety limits being exceeded and subsequent fuel failures. The staff achieved technical resolution of this problem and closed out generic safety issue #B19— Thermal Hydraulic Stability by issuing the Safety Evaluation Report (SER) on General Electric topical report NEDE-22277-P, "Compliance of General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria." This SER, submitted as Amendment 8 to General Electric Standard Application for Reactor Fuel (GESTAR), concludes that operating limitations which provide for the detection and suppression of flux oscillations in operating regions of potential instability, consistent with the recommendations of General Electric Service Information Letter 380, are acceptable to demonstrate compliance with General Design Criteria GDC-10 and GDC-12, for cores loaded with approved GE fuel designs.

The resolution of this issue will have a beneficial effect on all operating BWRs, since the review of BWR single loop operation (SLO) has been complicated by potential thermal-hydraulic (T-H) instability and jet pump vibration problems. In low-flow operating regions, it has been necessary to develop special operating procedures to assure that General Design Criteria 10 and 12 are satisfied in regard to thermal-hydraulic instabilities. This has resulted in BWRs being limited to 24 hours of operation and/or being limited to less than 50 percent power levels during SLO. With the resolution of the thermal-hydraulic stability issue, restrictions such as the 50 percent power limitation during SLO will be lifted and all BWRs which properly monitor and avoid regions of T-H instabilities will now be allowed to operate permanently with a single loop out of service. Thus, the resolution of the thermal-hydraulic stability issue was a major factor in allowing the staff to close generic safety issue #B-59, (N-1) Loop Operation in BWRs and PWRs. The resolution of the thermal-hydraulic stability issue will lead to safer operation for all of the U.S. BWRs and to economic benefits for BWR owners through increased power output.

Transamerica Delaval Diesel Generators

During a load test on August 12, 1983, the main crankshaft failed on one of the three emergency diesel generators (EDGs) at the Shoreham Nuclear Power Station (N.Y.). The EDGs at Shoreham were manufactured by Transamerica Delaval, Inc. (TDI), which has supplied 54 EDGs to 14 other nuclear power plant sites in the United States.

While evaluating the Shoreham failure, the NRC staff and the industry became aware of a broad pattern of deficiencies involving critical engine components in TDI EDGs at Shoreham and other nuclear and non-nuclear facilities. These deficiencies appeared to stem from inadequacies in design, manufacture, and quality assurance/quality control on the part of TDI. In addition to crankshafts, problem areas have included an engine block failure, piston failures, cracked and leaking cylinder heads, excessively worn turbocharger thrust bearings, and rupture of a defective fuel line. In response to these problems, 13 nuclear utilities formed an Owners Group to establish a program for upgrading and confirming the adequacy of the TDI diesels for nuclear service.

Phases I and II of the Owners Group program were essentially completed in fiscal years 1984 and 1985, respectively. Under Phase I, the Owners Group developed a proposed technical resolution to known significant problem areas to serve as a basis for plant licensing during the period prior to completion of Phase II.

Phase II of the Owners Group Program has proceeded beyond known problem areas to consider systematically all components (approximately 150-to-170 component types per engine) important to the operability and reliability of the engines. Phase II is intended primarily to ensure that significant new problem areas do not develop in the future because of deficiencies in design or quality of manufacture. The Owners Group performed the Phase II design reviews and recommended needed component upgrades, modifications and inspections to validate the quality of manufacture and assembly. The preparation of a comprehensive engine maintenance and surveillance program to be implemented by the individual owners is a major element of the Phase II Program.

The staff expected to have completed its final evaluation of the Owners Group findings and recommendations stemming from this program in the fall of 1985. In the interim, the staff concluded that issues warranting priority attention have been adequately resolved at several plants and that the TDI EDGs will provide reliable service through at least the first refueling outage (by which time the staff will have completed its overall review). This finding has permitted the staff to proceed with issuance of operating licenses for these plants because of (1) actions taken by the Owners Group and the individual owners to resolve known problem areas, (2) implementation of an acceptable engine maintenance and surveillance program, and (3) incorporation of plant Technical Specification requirements and operating procedures that ensure the engines will not be operated in an overstressed condition.

During fiscal year 1985, Supplemental Safety Evaluation Reports concerning TDI diesel generators were issued to support issuance of Operating Licenses for Shoreham Nuclear Power Station Unit 1 (N.Y.), Catawba Nuclear Station Unit 1 (S.C.), and River Bend Station Unit 1 (La.). In addition, a Safety Evaluation Report (SER) was issued to support operation of San Onofre Nuclear Generating Station Unit 1 (Cal.) until its next refueling outage. SERs addressing long term resolution of TDI engines issues will be issued prior to restart from the first and/or next refueling outages at the above plants.

Hearings before the Atomic Safety and Licensing Board (ASLB) on the subject of the TDI engines at Shoreham Nuclear Power Station Unit 1 began on September 10, 1984, and ended on March 12, 1985. The board found there is reasonable assurance that the TDI diesels can perform their required safety function for the first refueling cycle. An ASLB also conducted hearings on April 10 and 11, 1985, regarding the TDI diesels at Perry Nuclear Power Plant Unit 1 (Ohio); all contentions were dismissed as a result.

Pipe Cracks at Boiling Water Reactors

Intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel piping in boiling water reactors (BWRs) has been observed for many years; however, extensive IGSCC in large-diameter recirculation system piping was reported for only the first time in the United States at Nine Mile Point Unit 1 (N.Y.) in March 1982. To resolve this concern of cracking in large diameter piping, the staff issued Inspection and Enforcement Bulletins 82-03 and 83-02 in October 1982 and March 1983, respectively, requiring augmented piping inspection of operating BWRs.

After extensive IGSCC was reported in several operating BWRs, Orders were issued to five operating BWR licensees to accelerate the inspection schedules for their facilities. The inspection revealed cracking in the welds of large diameter piping of both recirculation and residual heat removal systems at all operating BWR plants except Oyster Creek (N.J.), Big Rock Point (Mich.), Duane Arnold (Iowa), Millstone Unit 1 (Conn.) and Browns Ferry Unit 3 (Ala.).

The NRC staff discussed its short term approach for assuring continued safe operation of affected BWRs in SECY-83-267C. These staff short term reinspection and repair criteria, as modified by the Advisory Committee on Reactor Safeguards comments, were issued on April 19, 1984, in Generic Letter 84-11 to all licensees of BWR facilities for use in piping inspection and repair, subsequent to the inspections required by IE Bulletin 82-03, 83-02, and the five Confirmatory Order inspections. Approximately seven plants, including Duane Arnold and Millstone Unit 1, have completed reinspection. Many cracked piping welds were reported in each reinspected plant, including those previously reporting no cracking under Bulletin 82-03 or 83-02 inspections.

To facilitate piping replacement, the NRC issued Generic Letter 84-07 on March 14, 1984, providing procedural guidance to BWR licensees for piping replacement under 10 CFR 50.59. The guidance covers the engineering design, materials, fabrication and installation of replacement piping. Nine Mile Point Unit 1 (N.Y.) , Monticello (Minn.), Pilgrim Unit 1 (Mass.), Hatch Unit 2 (Ga.), and Cooper (Neb.) have completed piping replacement. Vermont Yankee and Dresden Unit 3 (Ill.) will replace piping during their next refueling outage.

The ultrasonic reinspection results reported by the utilities during the report period showed that the performance by the qualified ultrasonic testing (UT) personnel lacks uniformity and consistency. To resolve this concern, NRC has determined that all UT personnel performing detection and evaluation should be requalified to an upgraded program at EPRI/NDE Center. The test samples in the upgraded program consist of cracked field samples from a number of operating BWR plants, in addition to the original set from Nine Mile Point.

The NRC long term technical position on BWR pipe cracking, developed by the Task Group on Pipe Cracking under the auspices of the NRC Piping Review Committee, was published in NUREG-1061, Volume 1. The Task Group report concludes that the IGSCC in large diameter piping in BWR plants is not a new phenomenon; however, it is a serious problem requiring some changes in current regulatory practice. The report recommends that the remedies should consist of measures to combat all three causative factors: (1) the degree of sensitization in the materials, (2) tensile stresses in the piping and (3) conducive environmental conditions. Curing at least two of the three causative factors should be fully effective. The recommended schedule for augmented inspection for welds should depend on the degree of material resistance to IGSCC and the effectiveness of the mitigating processes used to reduce the susceptibility to IGSCC.

The NRC staff long-range goal as detailed in SECY-84-301 is to bring all plants to a condition that allows them to be inspected at frequencies specified in 10 CFR 50.55a(g) without relying upon the augmented inservice inspections. To meet this long-range goal, the Task Group report recommends that all piping should be made of IGSCC resistant materials, or uncracked nonresistant materials if the residual tensile stresses in the weld have been eliminated by either induction heating stress improvement or other means judged to be fully effective, and the reactor water chemistry environment has been modified by hydrogen additions to further reduce the potential for cracking.

To implement this long-range plan, the draft NUREG-0313 Revision 2, which incorporated the recommendations made by the Pipe Crack Task Group in NUREG-1061, Volume 1, has been completed. The draft NUREG-0313, Revision 2 was to issued for internal and public comments in October 1985. After resolving the internal and public comments, a generic letter incorporating this implementation document will be sent to all BWR licensees requesting their proposals for bringing their plant(s) into compliance with 10 CFR 50.55a(g).

Steam Generators

Degradation of steam generators (SGs) manufactured by each of the three pressurized water reactor (PWR) vendors has resulted from a combination of problems related to mechanical design, materials selection, fabrication techniques, and secondary system design and operation. The different forms of steam generator tube degradation identified include stress corrosion cracking, wastage, intergranular attack, denting, erosion-corrosion, fatigue cracking, pitting, fretting, support plate degradation, and mechanical damage resulting from impingement of foreign objects or loose parts on internal components. One or more of these forms of degradation have affected at least 40 operating PWRs and have resulted in extensive SG inspections, tube plugging, repair, or replacement. The staff provided a detailed description of steam generator tube operating experience in NUREG-0886 (February 1982) and NUREG-1063 (June 1985).

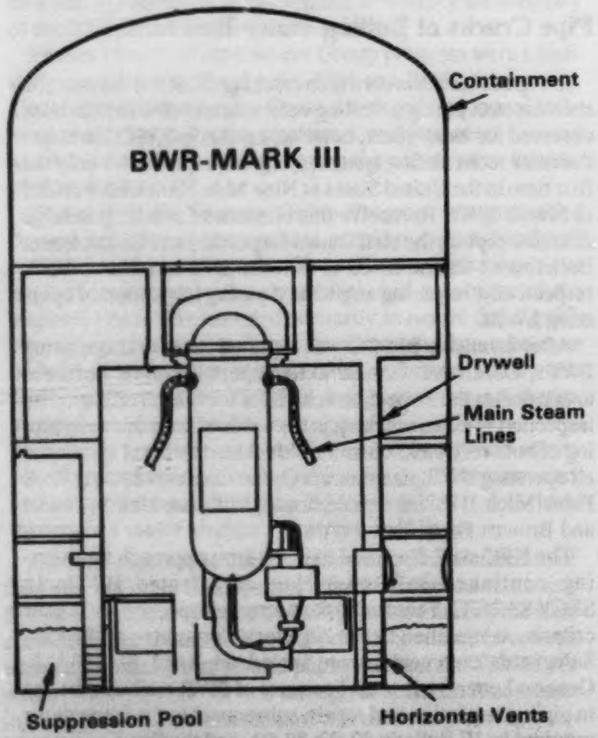
The majority of the SG tube failures that have occurred under normal operating conditions were small stable leaks. Some required plant shutdown, inspection, and corrective actions; but others were small enough (e.g., below the leak rate limit of the Technical Specifications) to permit plant operation until a scheduled shutdown. However, four significant SG tube ruptures have occurred in domestic PWRs since 1975; at Point Beach Unit 1 (Wis.) on February 26, 1975; Surry Unit 2 (Va.) on September 15, 1976; Prairie Island Unit 1 (Minn.) on October 2, 1979; and R. E. Ginna (N.Y.) on January 25, 1982.

Staff concerns stem from the fact that the tubes are a part of the reactor coolant system (RCS) boundary and that tube failures result in a loss of primary coolant. Further, tube failures allow primary coolant into the steam generators where isolation from the environment is not fully ensured.

Steam generator tube integrity was designated an unresolved safety issue (USI) in 1978 and Task Action Plans (TAP) A-3, A-4, and A-5 were established to evaluate the safety significance of degradation in Westinghouse, Combustion Engineering and Babcock & Wilcox steam generators, respectively. These studies were later combined into one effort because many of the problems being experienced were common to or similar for all the PWR vendors.

In May 1982, subsequent to the issuance of NUREG-0909, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," April 1982, the staff initiated an integrated program to consider the lessons learned from the Ginna steam generator tube rupture (SGTR) and from the three previous domestic SGTR events (NUREG-0651, "Evaluation of Steam Tube Rupture Events," March 1980), and to consider the recommendations on corrosion related degradation mechanisms. The recently issued NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4 and A-5 Regarding Steam Generator Tube Integrity," April 1985, addresses issues within the areas of steam generator integrity, plant systems response, human factors, radiological consequences and response to SGTR. A generic risk assessment indicates that risk from SGTR events is not a significant contributor to total risk at a given site, nor to the total risk to which the public in general is exposed.

The generic risk estimates notwithstanding, degradation of steam generator tubing has become widespread throughout the industry; the average age of the steam generators is still less than a third of the designed lifespan. Necessary repairs have caused high radiological exposure to workers and now contribute about 20 percent of the average annual plant occupational dose. SGTR events also represent a significant challenge to the plant operators since a variety of diagnoses and manual actions must be taken in a relatively short time. The report therefore identifies a number of actions which would reduce the incidence of steam generator tube degradation and the frequency of tube ruptures, and would mitigate the consequences of SGTR events. These actions would also further reduce risk and have been designated as "staff recommended actions." As part of the technical resolution of the unresolved safety issues, the staff has issued Generic Letter 85-02 to all pressurized water reactor licensees and applicants to inform them of the staff recommended actions and to request that they submit descriptions of their overall programs to ensure steam generator tube integrity and SGTR mitigation. The staff is evaluating these responses and will report its findings to the Commission in the near future.



In the Mark III containment for a boiling water reactor, steam escaping from a break in a main steam line would be condensed in a suppression pool to avoid increased pressure.

Table 10. Environmental Impact Statements Issued in FY 1985

Site	*Document*	*NUREG No.*	*Publication Date*
WNP-3	FES	1033	5/85
Hope Creek-1	FES	1074	12/84
Millstone-3	FES	1064	12/84
River Bend-1&2	FES	1073	1/85
Nine Mile Point-2	FES	1085	5/85
Beaver Valley-2	DES & FES	1094	12/84 & 9/85
Vogtle-1&2	DES & FES	1087	10/84 & 3/85

MARK III Containment

In a letter dated May 8, 1982, to the Mississippi Power and Light Company, a former lead systems engineer responsible for containment design at General Electric Company raised certain concerns related to the safety adequacy of the Mark III containment design. The staff asked the Mark III Owners Group, representing four plants, to not only respond to the individual's concerns, but also to set up an independent outside panel to review the Owners Group responses. This panel, called the Containment Issue Review Panel, issued its report on July 3, 1984. In late 1982, the NRC staff also contracted the Brookhaven National Laboratory to assist in the review of the individual Mark III owner responses. The staff and its contractor have completed this review and presented its evaluation in each plant's Safety Evaluation Report. In general, the staff's evaluation revealed that all but two of the individual's concerns have been addressed satisfactorily in the plant Safety Analysis Reports. These two areas include the Residual Heat Removal heat exchanger relief line load definition and the Safety Relief Valve discharge sleeve steam condensation load definition. The staff expected to complete its evaluation of these two issues by the end of the 1985. However, to assure safe operation, the staff imposed a license condition on two of the plants requiring them not to use the steam-condensing mode until sufficient justification is provided to alleviate the remaining safety concern.

Protecting the Environment

Environmental Impact Assessment

NRC staff prepared several environmental impact reviews related to operating license applications during fiscal year 1985. Final Environmental Statements (FES) were completed on five nuclear generating sites for which draft statements had been issued during fiscal year 1984 (see Chapter 2 of the *1984 NRC Annual Report*) Those sites were WNP-3 (Wash.), Hope Creek Unit 1 (Del.), Millstone Unit 3 (Conn.), River Bend Units 1 and 2 (La.), and Nine Mile Point Unit 2 (N.Y.). The staff also prepared Draft and Final Environmental Statements on two sites, Beaver Valley Unit 2 (Pa.) and Vogtle Unit 1 and 2 (Ga.), and initiated the operating-license stage environmental review on a third site, South Texas Units 1 and 2, for which a draft statement will be issued during fiscal year 1986.

The environmental reviews performed on the Beaver Valley and Vogtle sites identified only a few potential issues. The potential exists for annoyance-level noise impacts to a cluster of private residences from operations of the Beaver Valley station. The FES recommended that a short term noise monitoring program be conducted after operation begins, and that actual impacts and the need for mitigation be evaluated based on the program's findings.

The staff identified a high potential for bio-fouling of station water systems by Asiatic clams (species *Corbicula*) at the Vogtle plant, based on large clam populations in the Savannah River near the site and on clam infestations found at other nearby water intake systems. Bio-fouling at Vogtle will be controlled by chlorinating the cooling water. The staff evaluated the impacts to the river ecosystem from chlorine discharges in the FES and predicted they would be insignificant. Table 10 lists environmental impact statements issued during fiscal year 1985.

Transmission Corridor Crossing of a National Natural Landmark and State Scenic River

As part of the review for the Final Environmental Statement related to the operation of the Vogtle Electric Generating Plant Units 1 and 2 (Ga.), the staff discovered that one of the proposed transmission corridors was to cross Ebenezer Creek in

This portion of Ebenezer Creek in Burke County, Ga., is part of National Natural Landmark/Scenic River area, preserving one of the best remaining examples of a Cypress-Gum forest. An original proposal by owners of the Vogtle nuclear power plant near Waynesboro, Ga., called for clear-cutting a 150-foot wide transmission line path through this swamp, but it was changed after the NRC—in consultation with the U.S. National Park Service, Fish and Wildlife Service and Army Corps of Engineers—suggested the use of three very tall transmission towers, obviating the need for most of the planned clear-cutting.

Georgia. The point of crossing had previously been designated by the U. S. National Park Service as a National Natural Landmark and by the Georgia State Legislature as a Scenic River, because it is the best remaining example of a Cypress-Gum forest in the Savannah River Basin. As originally proposed, the transmission line would have created a 150-foot wide clear-cut corridor across the swamp. On being informed of the undesirable consequences of such a corridor, the utility suggested several less damaging proposals. Through the staff's consulation with the National Park Service, U. S. Fish and Wildlife Service and the Army Corps of Engineers, it was finally agreed that the most desirable alternative was to stop clear-cutting the rights-of-way at a line 2,675 feet north of the south bluff of Ebenezer Creek and then to construct three extra-tall towers—one to be installed on the south bluff of the creek, one 1,475 feet north of this tower and the third at the end of the clear-cut area. Because of the great height of the towers, no trees would have to be topped and only an area 100 x 100 feet would have to be cleared within the swamp, thus preserving the integrity of this unique area.

Environmental Responsibilities For Cancelled Projects

More than 100 commercial nuclear power projects for which license applications were filed have been cancelled. The more recent cancellations have occurred at a stage of the project where site preparation or major construction activities were under way. Large areas of the sites had been stripped of vegetation, subjected to earth-moving activities, excavation, and/or exposed to the forces of erosion. Some of the projects were actually more than 50 percent complete when cancelled.

The staff assessment of environmental impact done in conjunction with the initial license application review anticipated that construction would be completed in a timely manner; that the impacts associated with intermediate stages of construction would be temporary; and that they would be eliminated upon final grading and landscaping of the completed project. Thus, cancelling the projects in the middle of construction posed unexpected environmental issues requiring regulatory action.

The staff determined that under the National Environmental Policy Act a license application should not be returned to a cancelling utility while its site is in a condition which affects any adjoining areas. Although there would be no requirement for restoring a site to its pre-application conditions, the utility would be required to stabilize the site by grading and reseeding it, to control runoff that could damage streams and other nearby water-bodies by siltation. Although NRC does not have an interest in the use of the site after cancellation, the utilities' plans or options have, in some instances, had bearing on their scheduling of the necessary site stabilization measures.

Antitrust Activities

As required by law since December 1970, the staff has conducted prelicensing antitrust reviews of all construction permit applications for nuclear power plants and certain other commercial nuclear facilities. In addition, applications for amendments to construction permits or operating licenses that transfer an ownership interest in a nuclear facility to one or more additional applicants are subject to antitrust review.

An application for an operating license is not subject to formal antitrust review unless the NRC first determines that "significant changes" in the applicant's activities have occurred since the review of the application for a construction permit (see 47 FR 9983 for procedures used). During fiscal year 1985, four analyses were completed for determination of significant changes. In each instance, the finding was that the changes that had occurred were not significant in an antitrust context.

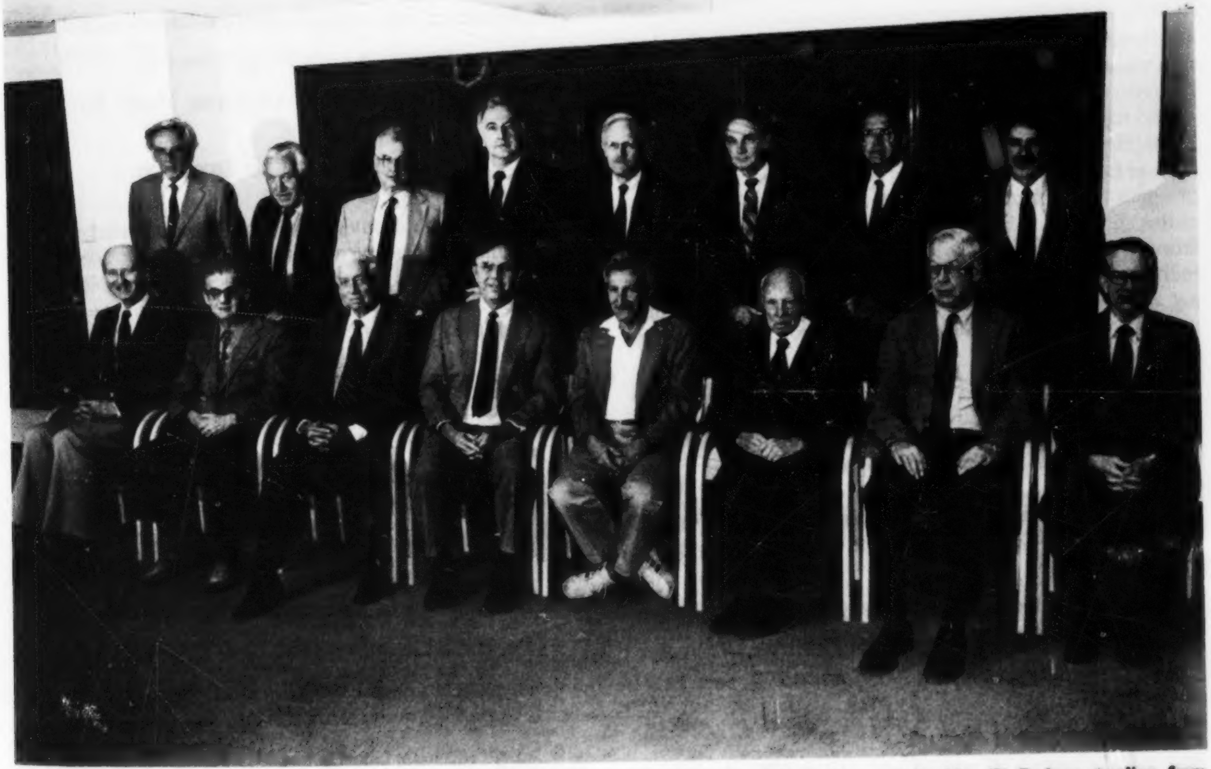
Remedies to antitrust problems usually take the form of conditions attached to licenses, and the NRC has the responsibility to enforce compliance with these antitrust conditions. During fiscal year 1985, the staff closed out enforcement actions pertaining to six of nine counts of a request for enforcement of antitrust conditions for the Diablo Canyon (Cal.) nuclear plant. The other three counts for Diablo Canyon, and 10 counts of a request for enforcement for an antitrust condition for the Farley (Ala.) nuclear plant, were still under consideration as of September 30, 1985.

In fiscal year 1985, the NRC staff documented its antitrust review and enforcement procedures by publication of NUREG-0970, "Procedures For Meeting NRC Antitrust Responsibilities."

Advisory Committee on Reactor Safeguards

The Advisory Committee on Reactor Safeguards (ACRS), established by statute in 1957, provides advice to the Commission on potential hazards of proposed or existing reactor facilities and on the adequacy of proposed safety standards. The Atomic Energy Act of 1954 also requires that the ACRS advise the Commission with respect to the safety of operating reactors and perform such other duties as the Commission may request. Consistent with the Energy Reorganization Act of 1974, the Committee will review any matter related to the safety of nuclear facilities specifically requested by the Department of Energy (DOE). Also, in accordance with Public Law 95-209, the ACRS is required to prepare an annual report to the U.S. Congress on the NRC Safety Research Program.

The ACRS reviews requests for preapplication site and standard plant approvals, each application for a construction permit or an operating license for power reactors, applications



The NRC's Advisory Committee on Reactor Safeguards (ACRS) held its 300th meeting in April 1985, when this photo was taken. Members of the ACRS are, as seated, from left to right, Dr. Dade W. Moeller, Dr. Chester P. Siess, Mr. Jesse C. Ebersole, Mr. David A. Ward (Chairman), Dr. Harold W. Lewis (Vice Chairman), Mr. Harold Etherington (member emeritus), Dr. William Kerr, and Dr. Max W. Carbon; standing, from left to right, are Dr. Robert C. Axtmann, Dr. J. Carson Mark, Mr. Glenn A. Reed, Dr. Forrest J. Remick, Dr. Paul G. Shewmon, Mr. Charles J. Wylie, Mr. Carlyle Michelson, and Dr. David Okrent.

The ACRS maintains a felllowship program enlisting graduate and postdoctoral nuclear scientists and engineers to assist in the committee's work. Shown above ACRS Senior Fellow John A. MacEvoy confers with ACRS Chairman David A. Ward, at right.

for licenses to construct or operate test reactors, spent fuel reprocessing plants and waste disposal facilities.

Consistent with the statutory charter of the Committee, all ACRS reports, except for classified reports, are made part of the public record. Activities of the Committee are conducted in accordance with the Federal Advisory Committee Act which provides for public attendance at and participation in Committee meetings. The ACRS membership, which is drawn from scientific and engineering disciplines, includes individuals experienced in chemistry and chemical engineering, electrical engineering, mechanical engineering, structural engineering, reactor operations, reactor physics and environmental health.

During fiscal year 1985, the Committee completed its annual report to Congress on the NRC Safety Research Program for fiscal years 1986 and 1987, and its annual report to the Commission on the Safety Research Program and Budget for fiscal year 1987.

The Committee also provided special topical reports to the NRC, individual Commissioners and others on a variety of issues, including:

- Proprietary Probabilistic Risk Analysis for Future Final Design Approval on a Proposed Standard Plant.
- Questions Regarding the Proposed NRC Severe Accident Policy.
- Control Room Habitability.
- Use of the "Check-Operator" Concept for Licensed Reactor Operator Requalification.
- Need for Information Concerning Preferred Operator Action for Postulated Scenarios in which a.c. Power is Restored After Severe Core Damage has Occurred.
- Possibility of an Organization like the National Transportation Safety Board (NTSB) for Nuclear Safety.
- A "Base" Program of NRC Safety Research.
- Notification of NRC of Significant Results Arising from Industry-Sponsored Probabilistic Risk Assessments.
- Comments on NRC Programs for the Quantification of Seismic Design Margins.
- National Academy of Sciences Study of Human Factors Research Needs in Nuclear Regulatory Research.
- Consideration of Earthquakes in Off-site Emergency Planning.
- Provisions for Protection Against Sabotage.
- Incident Investigation at the Davis-Besse Nuclear Power Station.
- ACRS Comments on Primary Coolant System Defect Evaluation.
- Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees.
- Nuclear Power Plant Control Room Habitability.
- Status Report on Long Range Planning.
- Systematic Analysis of Operating Plants for Severe Accidents.
- INPO Program on Radiation Protection.

The Committee's activities during the report period reflected the continuing licensing activity within the Commission and included six reports on requests for nuclear power plant operating licenses and one review of an operating plant evaluated as part of the systematic evaluation program.

In addition to its reports on licensed reactors and operating license applications, the Committee provided advice to NRC on proposed rules, criteria, or regulatory guides, including:

- Proposed Amendments to 10 CFR 20 to Specify Residual Radioactive Contamination Limits.
- NRC Proposed Final Rule on Pressurized Thermal Shock Events.
- Identification of New Unresolved Safety Issues.
- Backfitting Requirements.
- Regulatory Guide 1.28, Revision 3, "Quality Assurance Program Requirements (Design and Construction)."
- Proposed Rule Change to 10 CFR Part 50 Appendix E; Deletion of the Unusual Event Emergency Classification.
- Proposed Rule on Backfitting.
- NRC Staff Proposal for the Resolution of USI A-44, "Station Blackout."

- Proposed Amendments to 10 CFR 60, "Disposal of High-Level Radioactive Waste in Geologic Repositories."
- Proposed Regulatory Policy for Advanced Reactors.
- Severe Accident Policy—Systematic Review of Nuclear Power Plants
- ACRS Action on Proposed Regulatory Guide, Task No. IC 127-5.
- ACRS Action on the Proposed Revisions to Appendix J of 10 CFR 50 and the Related Regulatory Guide.
- Proposed NRC Safety Goal Evaluation Report.
- NRC Maintenance and Surveillance Program Plan.
- Status of USI A-46 (Seismic Qualification of Equipment in Operating Plants).
- Proposed Resolution for USI A-43, "Containment Emergency Sump Performance" and Regulatory Guide 1.82, Revision 1, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident."
- ACRS Action of the Proposed Revision 2 to Regulatory Guide 1.99.

The Committee commented in three reports on the NRC Staff's proposed priority rankings for newly identified generic issues.

The Committee also prepared four reports on events of particular interest at operating reactors and three reports dealing with various aspects of the NRC's civilian radwaste program.

In accordance with the procedures embodied in NRC Manual Chapter 4125, the Committee reviewed a Differing Professional Opinion concerning the proposed final rule on pressurized thermal shock.

In performing the reviews and preparing the reports cited above, the ACRS held 12 full Committee meetings and 104 subcommittee and working group meetings. Members of the Committee also participated in several conferences and visits to exchange safety-related information with foreign groups engaged in nuclear regulatory and development activities.

The ACRS met with the Groupe Permanent Reacteur in Paris, France, on May 20-22, 1985 and with the Reaktor-Sicherheitskommission in Munich, Federal Republic of Germany, on May 23-24, 1985. During these meetings specific items concerning safety, availability and reliability of standardized and next generation plants, improved valves, containment integrity, control room design and accident procedures, primary system blowdown on PWRs, operational experience of current plants and severe accident management were discussed.

[illegible]

- [illegible] Probabilistic Risk [illegible]
- Proposal to [illegible] the Proposed NRC Safety Goal [illegible]
- [illegible]
- Use of the "Leak-Before-Break" Concept [illegible] the Design of Nuclear Reactors.
- Need for Independent [illegible] Operating [illegible]

[illegible]

- Proposed Amendments to 10 CFR [illegible] Emergency Core Cooling [illegible]
- NRC [illegible]
- [illegible]
- Backfitting [illegible]
- Regulatory Guide 1.28, Revision 3, "Quality Assurance Program Requirements (Design and Construction)."
- Proposed Rule Change to 10 CFR Part 50 Appendix E, [illegible] Emergency Planning [illegible]
- Proposed Rule on [illegible]
- NRC Staff Proposal for the Resolution of USI A-[illegible], "Station Blackout."

CHAPTER 3

Cleanup at Three Mile Island

Substantial progress continued during fiscal year 1985 in all phases of the cleanup of the damaged Unit 2 reactor at the Three Mile Island Nuclear Power Station (TMI) near Harrisburg, Pa.

The safe removal and storage of the reactor vessel plenum assembly (PA) in May 1985 provided the access to the damaged reactor core necessary for the installation and operation of specially designed defueling equipment. Although delays in fabrication and delivery of this unique defueling equipment delayed the scheduled commencement of fuel removal activities from July until November of 1985, General Public Utilities Nuclear Corporation (GPU) still projects completion of the cleanup by mid-1988.

During fiscal year 1985, the highly radioactive reactor building basement was inspected through the use of a robotic vehicle. The PA was raised on jacks and all remaining attached fuel assemblies were dislodged prior to eventual PA removal and storage. Video inspections of the reactor vessel lower head revealed the distribution of core debris in that region and provided useful information for defueling planning and for revising previous theories regarding the accident. Decontamination and dose reduction activities continued in support of extensive defueling preparations. The processing and shipping of radioactive wastes also continued.

The cleanup funding situation continued to improve in fiscal year 1985, as GPU received payments from all sources that had pledged to contribute to the cleanup. By October 1985, the Edison Electric Institute, with support from six utilities in Pennsylvania and New Jersey, had paid GPU nearly $24 million of the $25 million pledged for calendar year 1985. The restart of TMI Unit 1 in October 1985 could result in an additional $15 million annual contribution to the cleanup from existing customer revenues. The financial aspects of the cleanup are addressed in more detail in Chapter 9.

Reactor Building Activities

A total of 238 entries were made into the TMI-2 reactor building during fiscal year 1985. First quarter activities included the inspection and jacking of the reactor vessel plenum, inspection of the polar crane, robotic inspection of the reactor building basement, and scabbling to reduce dose rates in work areas.

Based on indications that the plenum assembly had experienced deformation as a result of the accident, GPU elected to initially raise the PA on jacks to clear any potential interferences, in preparation for final lift. In December 1984, four hydraulic jacks were used to raise the 55-ton plenum assembly 71/2 inches. Long-handled tools were then used to detach remaining fuel assemblies and end fittings that adhered to the underside of the PA. Subsequent inspections indicated that all suspended debris had been dislodged and had fallen into the rubble pile in the core region.

During the fourth quarter of fiscal year 1984, GPU reported that one of the redundant brake systems on the reactor building polar crane had been found inoperable, because of a maladjustment of a manual brake release mechanism. The crane was initially removed from service and later its use was restricted to lifts of up to five tons. In January 1985, the polar crane was approved for full use, following an NRC inspection to verify the effectiveness of corrective actions taken by GPU. The polar crane was successfully used to lift and transfer the PA in May 1985.

Reactor building activities during the second quarter of fiscal year 1985 consisted of video inspections of the lower reactor vessel head, preparations for plenum assembly removal, and preparations for defueling, including partial installation of the Defueling Water Cleanup System (DWCS). Scabbling and water flushing activities were performed to further reduce reactor building radiation levels.

During the third quarter of fiscal year 1985, reactor building activities involved continued preparation for both plenum assembly transfer and for early defueling, including the installation of a dam in the fuel transfer canal, installation of additional DWCS components, assembly of fuel transfer equipment and installation of the defueling support structure. The 25-ton polar crane auxiliary hoist was load-tested and given its annual preventive maintenance. The PA was successfully transferred and stored during this period, as discussed below. Problems in the vendor's quality assurance program have resulted in delays in the delivery and acceptance of defueling canisters, canister storage racks and fuel transfer shields.

Reactor building activities during the fourth quarter of the fiscal year centered on early defueling preparations. The rotating defueling work platform with its cable management system was installed above the internals indexing fixture, directly over the open reactor vessel. Other defueling components installed included the service work platform, jib cranes, canister handling bridge, canister positioning system, defueling tool racks, and the fuel transfer mechanism. Installation of the vacuum defueling system began and the reactor building sump recirculation system was tested and declared operable. Additional inspections of the reactor vessel lower head were conducted as discussed below. Problems in the vendor's quality assurance program have resulted in delays in the delivery and acceptance of defueling canisters, canister storage racks and fuel transfer shields.

Plenum Assembly Transfer

The last major structural obstacle to defueling was removed in May 1985 when the plenum assembly was lifted from its jacked position in the reactor vessel, raised through the water-filled internals and transferred to its storage stand in the deep end of the fuel transfer canal. Prior to plenum transfer, a six-foot-high dam was constructed, allowing the deep end of the canal to be flooded to a level sufficient to provide adequate shielding for the stored plenum. The highest exposure rate during the transfer was 80 rems-per-hour at a point 3 feel below the plenum; the highest recorded exposure rate in the lead-shielded cubicle where workers were stationed was 30 millirems-per-hour.

The actual total occupational exposure for the PA transfer and storage operation was three person-rems, approximately 10 percent of the amount estimated in advance for this activity.

Reactor Vessel Lower Head Inspection

In February 1985, the first video inspection of the reactor vessel lower head region revealed the accumulation of a substantial quantity—estimated at 10-20 tons—of accident-generated debris. The debris bed had the appearance of a gravel pile composed of pieces nominally three-to-four inches long and half as wide. Similar material was observed by sighting up through the lower diffuser plate of the core support assembly. Although the composition of the debris could not be determined from the video inspections, it is evident that some molten material was generated during the accident, and that it resolidified and collected in the lower head area. Additional inspections conducted in July 1985, focusing on other quadrants in the lower head, disclosed that the debris bed was more shallow and individual pieces smaller in those areas, in contrast to the earlier determinations. In a separate effort, EG&G Idaho, Inc., under contract to the Department of Energy, ascertained that some areas of the core had reached temperatures of at least 5,100F (the melting point of uranium dioxide fuel) during the 1979 accident. This information, along with the lower head inspection data, will be used to revise certain theories of the TMI-2 accident sequence.

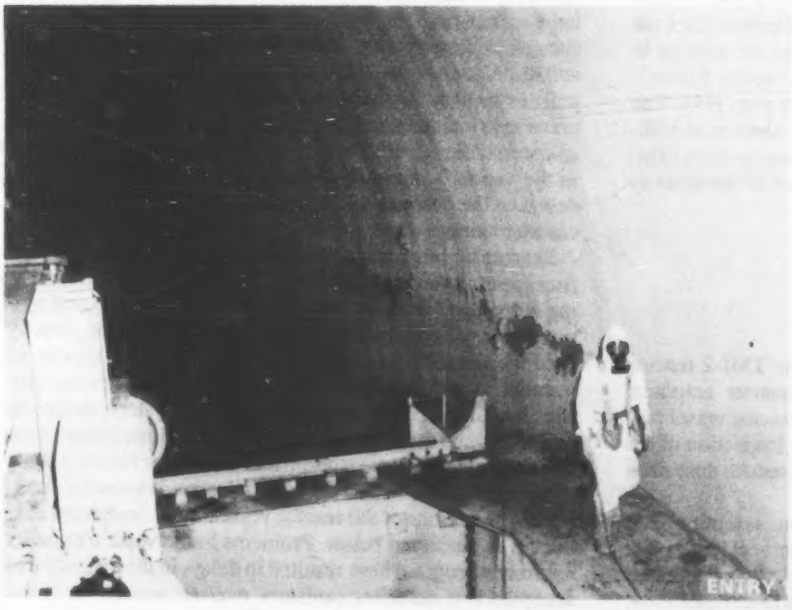
Waste Management

During the report period, the Submerged Demineralizer System (SDS) and the EPICOR-II system continued to be used to process radioactive water. The SDS was used to process reactor coolant, contaminated water generated from the makeup and purification demineralizer elution activities, reactor building sump water, and other water needing decontamination. The EPICOR-II system was also used to process miscellaneous waste water and to cleanse the effluent from the SDS. The SDS and EPICOR-II systems processed about 465,000 and 509,000 gallons of water, respectively, during the fiscal year. Two SDS liners and four EPICOR-II liners were shipped to the burial site at Richland, Wash.

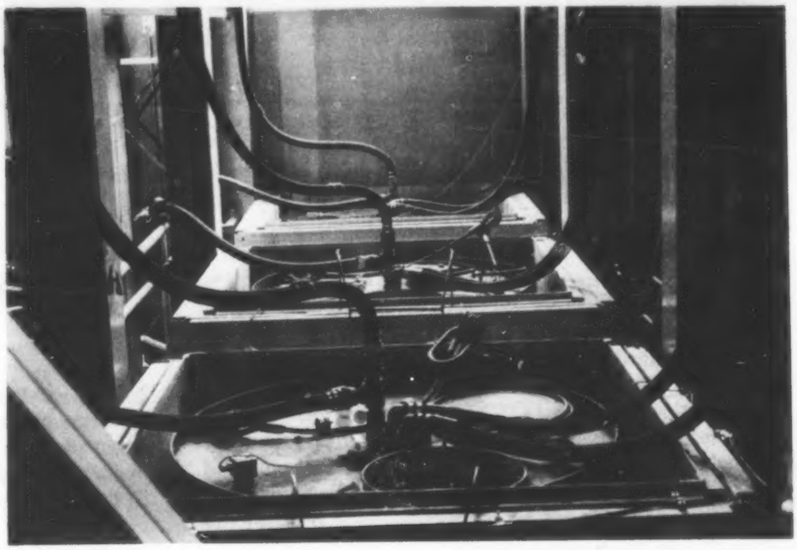
GPU Nuclear's burial privileges at the U.S. Ecology burial site in Richland were temporarily suspended in August 1985 when three barrels, out of a shipment of 104, were erroneously classified, labeled and certified by GPU as Class A radioactive waste. The privileges were restored after Washington State officials approved corrective measures taken by GPU to prevent future shipping and classification violations.

Decontamination and Dose Reduction Activities

Throughout fiscal year 1985, GPU continued decontamination and dose reduction activities aimed at maintaining



Some 238 separate entries into the reactor building at Three Mile Island Unit 2 occurred in 1985. Shown here is one of the early inspections of the plant's polar crane.



The Epicor II system to process radioactive water at TMI-2 continued in use during 1985 as part of the cleanup activity which has been going on since the accident in March 1979. The process vessels shown here are part of the Epicor system. They contain ion-exchange resins and are fitted with "quick-disconnect" hoses for liquid waste influent and processed waste effluent, with a vent line and overflow hose. Vented air from the vessels passes through special filter and charcoal adsorbers.

exposures to workers as low as reasonably achievable. Scabbling, a mechanical technique for removing the upper layer of concrete from a surface, was successfully employed in the reactor building and auxiliary and fuel handling building.

Upon completion of the bulk of reactor building scabbling activities in 1985, exposure rates on the entry level (305 feet elevation) and refueling floor (347 feet elevation) were reduced to 67 millirems-per-hour and 35 millirems-per-hour, respectively, a decrease of 30-70 percent. Shielding of the reactor building air coolers in conjunction with decontamination efforts and extensive pre-task training contributed to the lower-than-anticipated occupational exposure incurred during plenum assembly removal and transfer.

In the auxiliary and fuel handling building, scabbling and water flushes were used in the decontamination of the primary coolant makeup and letdown valve alleys, reactor coolant bleed tank rooms, the auxiliary building elevator, and various cubicles. A water flush of the seal return water system resulted in a 97 percent reduction in local exposure rates.

Chemical elution of the cesium from the highly radioactive makeup and purification demineralizer resins was completed in 1985. Approximately 4,200 curies of cesium-137 were removed from the resins, which were then placed in we layup in reactor coolant system quality water.

In November 1984, a robotic vehicle was used to inspect the highly radioactive building basement (282 feet elevation). General area radiation levels measured from 10 to 70 rems-per-hour, with hot spots as high as 1,100 rems-per-hour, confirming predictions that the basement is basically inaccessible to humans.

Advisory Panel on TMI Cleanup

The Advisory Panel for the Decontamination of Three Mile Island Unit 2, made up of citizens, scientists and local and State government officials, was formed by the NRC in 1980 to gain input from area residents regarding major cleanup activities. (See Appendix 2 for current membership.) In August 1985, the Commission approved a revision to the panel's charter to allow the panel to provide advice on the public's reactions to plans and results of certain health effects studies related to the TMI-2 accident. During the report period, the panel held eight public meeting in Harrisburg and Lancaster, Pa., and in Annapolis, Md., and met three times with the NRC Commissioners in Washington, D.C. Topics discussed by the panel during the year included TMI-2 health-effects studies, cleanup funding, flow of information to the panel, radiation protection issues, and NRC investigation and enforcement actions. The panel also received technical presentations on plenum assembly removal, Kr-85 monitoring during head lift, reactor vessel defueling, fuel shipping, and disposition of accident-generated water.

Operational Experience

CHAPTER 4

ANALYSIS AND EVALUATION OF OPERATIONAL DATA

NRC's Office for Analysis and Evaluation of Operational Data (AEOD) was established in 1979, several months after the accident at TMI-2. The Office, which reports directly to the Executive Director for Operations, is dedicated to the collection, assessment, and feedback of operational data to the NRC, the nuclear industry and the public.

The mission of the Office is to analyze and evaluate operational safety data associated with all NRC-licensed activities. These include the operations of commercial power reactor licensees and radioactive material and fuel cycle licensees. The Office also coordinates the overall NRC operational data program and serves as the focal point for interaction with outside and foreign organizations performing similar work.

Among the AEOD's specific activities are the following:

- Screen U.S. and foreign operational events for significance; systematically and independently analyze these events; seek trends and patterns that indicate potential safety problems; and develop and track AEOD recommendations for action by other NRC offices.
- Develop and coordinate agency guidance on Licensee Event Report requirements and monitor the effectiveness of the Nuclear Plant Reliability Data System.
- Develop and maintain computerized storage and retrieval systems for reactor and non-reactor operational data, including foreign data.
- Prepare and coordinate the quarterly Report to Congress on Abnormal Occurrences (NUREG-0090 series), as required by Section 208 of the Energy Reorganization Act of 1974; the bi-monthly Power Reactor Events reports (NUREG/BR-0051 series); the monthly Licensee Event Report Compilation (NUREG/CR-2000 series); and other feedback documents.
- Prepare reports of U.S. events for transmittal to the Nuclear Energy Agency's Incident Reporting System.
- Serve as the principal point of contact with the Advisory Committee on Reactor Safeguards (ACRS), the Institute of Nuclear Power Operations (INPO), and the Nuclear Safety Analysis Center (NSAC) on matters involving the collection and evaluation of operational data.

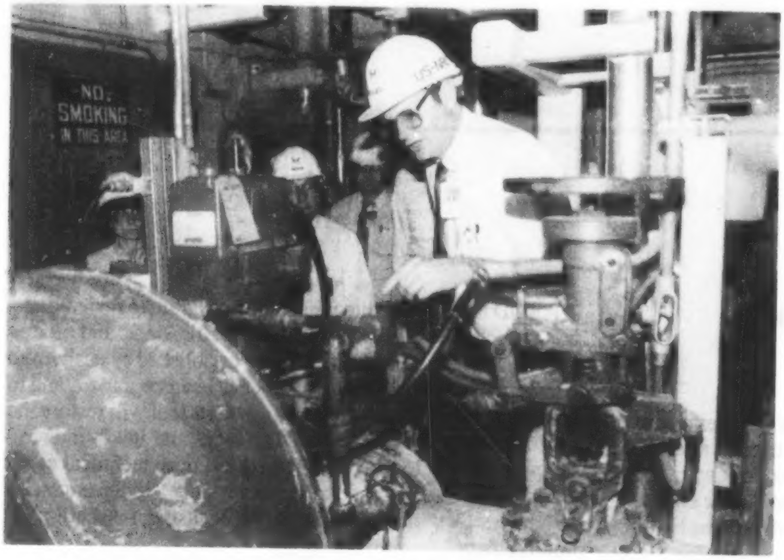
AEOD is part of an integrated NRC program to review operating experience in order to identify specific events and generic situations where the margin of safety established by design and confirmed in the licensing process has been degraded, and to identify and implement corrective actions that will restore the original margin of safety. AEOD's focus and involvement in the program are to provide a strong in-house technical capability in analysis of operating experience, independent of regulatory activities associated with licensing, inspection, or enforcement.

NRC Handling of Operational Data

Domestic. On January 1, 1984, a rule modifying and codifying the Licensee Event Report (LER) system became effective (10 CFR 50.73). The LER system had previously been defined in technical specification requirements. The new system permitted a more systematic analysis of operational events. Additional analysis is now provided on such issues as reactor scrams, emergency safety features actuations, and total system failures.

In fiscal year 1985, the Office continued to resolve questions on the interpretation of the new rule, and to conduct assessments of LER reporting. In September 1985, Supplement 2 to the original guidance document, NUREG-1022, was issued, providing an evaluation of first-year results and recommendations for improvement.

Under contract with the Nuclear Operations Analysis Center (NOAC) at Oak Ridge, Tenn., AEOD operates and maintains the Sequence Coding and Search System (SCSS), a computerized storage and retrieval system for LER data. Its objective is to encode all of the relevant technical information provided by the licensee in the LER with sufficient "tags" for the ready retrieval of individual items. During fiscal year 1985, about 2,700 LERs were added to the system, which is consistent with an estimate in 1983 that the number of LERs would probably decrease by 50 percent following implementation of the rule. This increased the number of LERs added to the data base (since 1981) to more than 16,500. Expansion of the data base to include LER data from 1980 will be completed in 1986. In April 1985, AEOD and the Oak Ridge National Laboratory published a set of manuals on the SCSS (NUREG/CR-3905), including the SCSS Coder's Manual, the list of SCSS codes, and Revision 1 to the SCSS User's Guide. During the report period, SCSS also was made directly accessible to over 40 users in NRC Headquarters and Regional Offices.


NO SMOKING IN THIS AREA

NRC resident inspector Walt Rogers explains the operation of an auxiliary feedwater pump, one of the two which failed in a June 9, 1985 incident at the Davis-Besse Nuclear Plant in Ohio. Rogers is shown pointing out a problem with the auxiliary feedwater turbine to Commissioner James Asselstine and NRC Region III Administrator James G. Keppler, during a tour of the plant following the mishap.

A "trends-and-patterns program" for analyzing LER data was initiated in 1984. The program uses statistical techniques to detect trends or patterns from incidents of low individual significance that may signify an unrecognized safety concern. The program encompasses the present SCSS and in the future will include the Nuclear Plant Reliability Data System (NPRDS), a voluntary, industry-run system for reporting failures of safety components. Implementation of the NPRDS by the industry's Institute of Nuclear Power Operations (INPO) is monitored by the NRC.

At the request of the Commission, a continuing NPRDS evaluation program is carried out by NRC staff. Two semiannual evaluation reports on NPRDS progress were forwarded in February 1985 (SECY-85-56) and August 1985 (SECY-85-56A). These reports noted that, while the system has substantially improved since 1981, the majority of nuclear power plants have not met the threshold considered representative of "active participation." The NRC staff is concerned that the NPRDS may not reflect a consistent and high level of reporting from all units—a prerequisite for the system to be usable as the basis for statistical analysis of component failures, and as a source of component failure data for specific analyses. These concerns will increase if significant improvement in the extent of participation, as predicted by INPO, is not achieved during the next assessment period.

Foreign. In fiscal year 1985, the NRC continued efforts to increase the number and usefulness of foreign experience reports that are received. The agency also participated in the exchange of operational event information with other countries through activities involving the Nuclear Energy Agency, the International Atomic Energy Agency, and various bilateral agreements. An NRC program at the NOAC systematically screens and assesses selected foreign information for its applicability to the U.S. program, and to abstract it for computerized data filing. This file now contains information on more than 5,500 foreign events.

Incident Investigation Program

During 1984 and 1985, the NRC sought—in response to a Congressional request—to identify potential improvements in the existing program for the investigation of significant operational events. Brookhaven National Laboratory was contracted by the NRC to perform a study to find ways to improve the program. As a result of that effort, the NRC staff identified a number of changes in the approach to investigating significant events. The most noteworthy change is the investigation of such incidents by a multi-disciplined Incident Investigation Team (IIT), made up of technical experts from the various NRC offices. These teams would prepare a single comprehensive report for each incident describing the event, setting forth the relevant facts, identifying root causes, and presenting findings and conclusions. The use of IITs, which would be administered by AEOD, is the focus of an expanded Incident Investigation Program.

ITT at Davis-Besse. On June 9, 1985, the Davis-Besse plant (Ohio) experienced a complete loss of main and auxiliary feedwater for about 12 minutes during an event involving an automatic shutdown from operation at 90 percent power. The event involved a number of equipment malfunctions and extensive operator activity, including actions outside the control room. Several operator errors occurred during the event. Because of the potential safety implications of this event, the NRC Executive Director for Operations appointed an IIT, consistent with the proposed Incident Investigation Program. The IIT collected and evaluated information to determine the sequence of operator, plant, and equipment responses during the event, and the causes of equipment malfunctions and operator errors. Problems identified included issues specific to Davis-Besse, as well as several possible generic issues. The underlying cause of the loss of main and auxiliary feedwater, as determined by the IIT, was the licensee's lack of attention to detail in the care of plant equipment. (The results of the IIT investigation of the Davis-Besse event are contained in NUREG-1154. The event is further described in Chapters 2 and 8.)

Analysis of Non-Reactor Operational Experience

In addition to the screening and analysis of reactor operational experience, the Office reviews the non-reactor operational experience associated with the activities and facilities licensed by the Office Of Nuclear Material Safety and Safeguards (see Chapter 5) and by Agreement States (see Chapter 9). AEOD also conducts studies from a human-factors perspective on both reactor and non-reactor operational events, and maintains and updates the non-reactor and medical misadministration data files. From the events screened, about 200 non-reactor events and 400 medical misadministrations are entered into the files each year. (See further discussion below of non-reactor data.)

Semiannual Report to the Commission

In April 1985, the Office submitted its second semiannual report to the Commission (AEOD/502) for the July-December 1984 period. Based on extensive screening, analysis and feedback of operational experience, AEOD registered the following comments and observations in its report:

(1) Analysis of 1984 LERs provides a broad overview of industry's operational experience, and indicates a tremendous range in the nature and number of reportable events by each plant.

(2) Licensee programs for operational experience assessment are diverse, and their characteristics and possible effectiveness vary widely. Resources are focused on the operating experience of their own facilities, and industry-wide experience reported from other plants may not be sufficiently emphasized at some plants.

(3) Based on data from events determined to be abnormal occurrences by the Commission, no favorable trend is evident industry performance with regard to the number of significant operational events in recent years.

(4) U.S. reactors continue to have frequent actuations of the reactor protection system, although the average rate of reactor scrams in 1984 decreased by about 9 percent from 1983 (from 6.5 to 5.9 per year per plant).

(5) Actuations of engineered safety features are frequent in a number of plants, yet most actuations are unnecessary.

The semiannual report presents supporting comments and data for these observations and provides summaries of Office activities during the report period. In the future, this report will be submitted annually in the spring.

ANALYSIS OF REACTOR OPERATIONAL EXPERIENCE

AEOD is responsible for screening LERs and other event-related documentation; identifying events of particular significance; conducting appropriate engineering evaluations and case studies for significant events; and formulating appropriate recommendations for action by other NRC offices. AEOD provides an in-house engineering capability for examining operational events at U.S. and foreign light-water commercial reactors. Its technical studies and evaluations are based upon U.S. and foreign event reports and supplemental information, and on the knowledge and experience of the technical staff. Reviews are normally initiated after a licensee report is available, so that Office activities are normally independent of and occur later than the prompt action that may be initiated by the Region or Program Offices to investigate an operating event and determine the need for immediate licensee response or generic action.

A concerted effort has been made to expand AEOD studies beyond the review of specific events and potential generic concerns, in order to encompass broad trend-and-pattern analysis of operational data. Further, the scope of AEOD efforts has been increased not only to look at the technical aspects of operational data—such as hardware and human performance—but also to focus more on other important aspects, such as how best to put the lessons of industry experience to use.

During fiscal year 1985, eight special studies and case studies (see Table 1), and nearly 30 engineering evaluations and technical reviews (see Tables 2 and 3), were completed by AEOD. Subjects examined in the evaluations and reviews included failures of motor-operated valves because of the "hammering effect" which occurs when these valves are subjected to repeated closure attempts, after a valve has already reached its fully closed position; failures of safety-related pumps because of debris in the emergency core cooling system; and complications resulting from the electrical interaction between units of the McGuire (N.C.) plant during a loss of off-site power in August 1984. Selected special studies and case studies on reactor operational experience are summarized below.

Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors

An AEOD engineering evaluation report issued in May 1984 dealt with a stuck-open testable isolation check valve on the low pressure coolant injection line at the Hatch Unit 2 (Ga.) plant. The assessment was that the safety significance of the event was high. This mispositioned valve substantially increased the likelihood of an interfacing loss-of-coolant accident involving the reactor coolant system (RCS) and the residual heat removal (RHR) system. Such an accident would involve the discharge of high pressure, high temperature reactor coolant into the low pressure RHR system and could result in a loss of integrity. Such a rupture would likely disable at least one train of the RHR system and would certainly bypass the containment. The isolation check valve had been held open by its attached air operator as a result of pneumatic pressure reversal caused by maintenance errors.

In August 1984, a testable isolation check valve of the core spray system at Browns Ferry Unit 1 (Ala.) failed in the open position, in an event almost identical to that discussed above. The mispositioned check valve, together with an inadverten opening of a normally closed motor-operated injection valve, led to an overpressurization of the core spray system. The core spray system was pressurized to near operating reactor pressure and temperature. Paint on sections of piping vaporized and actuated a smoke detector. Steam from the core spray system's relief valve, which opened to relieve the overpressurization, discharged into an open drain line. The mixture of water and steam which aprayed from the drain line contaminated 13 workers responding to the fire alarm.

Prompted by these events, and in order to judge the need for generic corrective actions, the Office broadened the scope of its search to include all testable isolation check valve failures in emergency core cooling and reactor core isolation cooling systems in boiling water reactors since 1975.

The review succeeded in identifying a total of eight events—including the events at Hatch and Browns Ferry—involving the failure of a testable isolation check valve, which provides the first isolation barrier between either the RCS or the feedwater system and an emergency core cooling system. Among these events, five involved an additional failure of the second and final isolation barrier, by means of an inadvertent opening of a normally closed motor-operated injection valve. Four of these five events occurred during power operation, thus leading to an overpressurization of an emergency core cooling system. A fifth event occurred while the plant was in cold shutdown, and so an overpressurization of the associated emergency core cooling system did not result; but a rapid draining of the reactor vessel did occur.

Among the eight observed failures of the testable isolation check valve (stuck open), five were associated with interference by the attached air operator, two involved causes related to the check valve itself, and one involved a failure whose cause remains unknown. All of the five observed filures of the normally closed motor-oerated injection valve involved inadvertent opening resulting from personnel errors committed during surveillance testing of the safety systesm.

Collectively, these operating events indicate a trend which has potentially serious safety implications and the likelihood that an interfacing loss-of-coolant accident is higher by two to several orders of magnitude than had been previously assessed.

The Tennessee Valley Authority's Browns Ferry nuclear plant near Decatur, Ala., was the site of an August 1984 incident in which an isolation test valve in the Unit 1 core spray system failed in the open position. This condition led to an overpressurization and a release of steam from the core spray system's relief valve, contaminating 13 workers in the area.

Table 1. AEOD Reports Issued During FY 1985

Case and Special Studies *Designation*	*Subject*	*Issued*
C501	Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants	6/85
C502	Overpressurization of Emergency Core Cooling Systems in Reactors	9/85
P501	Feedwater Transient Incidents in Westinghouse PWRs	7/85
P502	Trends and Patterns Analysis of 1981 Through 1983 LER Data (NUREG/CR-4129)	6/85
P503	Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Reactors January 1 Through June 30, 1984	8/85
P504	Trends and Pattern Report of Unplanned Trips at U.S. Light Water Reactors in 1984	8/85
S501	Review of Operational Experience from Non-Power Reactors	3/85
S502	AEOD Semiannual Report for July-December 1984	4/85

Recommendations developed by AEOD to minimize these occurrences include modifying the non-safety-related air operators and the affected testable isolation check valves, and reducing human error in maintenance and testing activities. These recommendations were under active review within the agency at the close of the report period. (See also the summary of "Degraded Isolation Valves in Emergency Core Cooling Systems," later in this chapter under "Abnormal Occurrences.")

Engineered Safety Features Actuations At Commercial U.S. Reactors

In order to gain a better understanding of the reasons for the frequency of challenges to safety systems in nuclear power plants, the Commission required that, effective January 1, 1984, actuation of any engineered safety feature (ESF) be reported to the NRC in an LER. Prior to that time, such actuations were not directly reportable. As a result of the revised requirement, and as part of the AEOD trends-and-patterns analysis program, a study was initiated of ESF actuations which occurred between January 1 and June 30, 1984.

The investigation was limited to those ESF actuations which occurred in systems other than the reactor protection system (RPS), which was the subject of a companion AEOD study. The objectives of this study were to gain an understanding of the frequency and causes of such ESF actuations on both an individual unit and an industry-wide basis; to determine the significance and implications of the current rate of actuations; to decide whether specific action by the NRC or the industry appeared warranted; and to investigate the usefulness of ESF actuation data as a valid indicator of licensee performance.

It was found that 501 ESF actuations occurred during the first six months of 1984, with 293 of the 1,195 LERs (25 percent) submitted describing at least one such ESF actuation. Of the 87 units issuing LERs, 61 units reported at least one ESF actuation. Twelve units reported more than 10 ESF actuations. The maximum number experienced at any one unit was 82. However, only 10 percent of all reported ESF actuations involved an emergency core cooling system, and none of those actuations was necessary to control an actual loss-of-coolant accident. About 70 percent of the actuations that occurred in ESF systems were associated with either an isolation function or a ventilation function.

Based on the analysis and evaluation in this report, AEOD concluded that, in general, the events necessitating ESF actuations have not been individually significant, and their occurrence frequency should not be a major concern. It is apparent, however, that the majority of the reported ESF actuations were unnecessary, and that their rate of occurrence could be greatly diminished by reducing the number of equipment failures during normal operation, reducing the number of personnel errors during maintenance and testing, and revising actuation setpoints to more appropriate protective levels.

Nine plants in the study were found to apparently be experiencing repeated unresolved actuations which could ultimately challenge continued equipment operability and proper

Table 2. Reactor Engineering Evaluations and Technical Reviews

Engineering Evaluation	*Subject*	*Issued*
E425	HPCI System Lockout at Vermont Yankee	10/11/84
E426	Single Failure Vulnerability of Power Operated Relief Valve Actuation Circuitry for Low Temperature Overpressure Protection	10/24/84
E427	Licensee Event Reports that Address Situations Which Potentially Could Result in Overloading Electrical Equipment in the Emergency Power System or Prevent Operation of the On-site Power System Sequencer	11/6/84
E501	Motor Operated Valve Failures Due to Hammering Problem	1/17/85
E502	Failure of RHR Suppression Pool Cooling Valve to Operate	1/25/85
E503	Partial Failures of Control Rod Systems to Scram	3/4/85
E504	Loss or Actuation of Various Safety-Related Equipment Due to Removal of Fuses or Opening of Circuit Breakers	3/29/85
E505	Service Water System Air Release Valve Failures	3/29/85
E506	Valve Stem Susceptibility to IGSCC Due to Improper Heat Treatment	5/13/85
E507	Electrical Interaction Between Units During Loss of Off-site Power Event of August 21, 1984 at McGuire 1 and 2	5/17/85
E508	Nuclear Plant Operating Experience Involving Safety System Disturbances Caused By Bumped ElectroMechanical Components	5/24/85
E509	Salem Unit 2 Depressurization Event	7/25/85
E510	Disabling of a Shared Diesel Generator Set Due to Electrical Power Supply Arrangement for Support Auxiliaries	7/30/85
E511	Closure of Emergency Core Cooling System Minimum Flow Valves	8/9/85
E512	Failure of Safety-Related Pumps Due to Debris	9/4/85
E513	High Pressure Core Spray System Relief Valve Failures	9/16/85
T423	Inoperability of Helium Circulatory Overspeed Trip Channels Due to Impedance Variations in Speed Sensing Cables Exposed to Steam Leak	10/25/84
T424	Fire Water Main Leakage Into 4 kV Switchgear Room at San Onofre 1	11/20/84
T501	Failure of Automatic Protection for Boron Dilution Event at Callaway Unit 1	1/22/85
T502	Comparative Analysis of Recent Feedline Waterhammer Events at Maine Yankee, Calvert Cliffs, Palisades, and Salem	3/18/85
T503	Pressurizer Level Instrumentation of Combustion Engineering Reactor Units	5/2/85

Technical Review	*Subject*	*Issued*
T504	Loss of Instrument Air and Subsequent Pressure Transient at Callaway 1	5/17/85
T505	Beaver Valley Component Cooling Water Pump Damage	7/17/85
T506	Primary System Release Due to Pressurizer Degas Relief Valve Lifting	7/25/85
T507	Standby Liquid Control System Pressure Relief Valves Lift at a Pressure Lower Than Reactor Coolant Pressure	8/13/85
T508	Browns Ferry Nuclear Plant HPCI System Performance Assessment	8/14/85
T509	Inadequate Surveillance Testing Procedures for Degraded Voltage and Undervoltage Relays Associated with 4160 Volt Emergency Buses	8/29/85
T510	Xenon Induced Power Oscillations at Catawba	9/4/85

personnel response. These plants are D.C. Cook Unit 2 (Mich.), Fort Calhoun (Neb.), LaSalle Units 1 and 2 (Ill.), San Onofre Units 2 and 3 (Cal.), Sequoyah Units 1 and 2 (Tenn.), and Washington Nuclear Unit 2. AEOD will continue to monitor these units to see if corrective actions are effective in resolving these actuations.

Further, AEOD found four potentially significant problems in the ESF actuations studied, including (1) improper temperature switch configuration, (2) steam supply transfer relay seal-in circuitry, (3) pressure switch location and setpoint calibration, and (4) component cooling water system interaction. AEOD will investigate these items to ascertain whether further actions, either generic or unit-specific, should be taken to properly address safety concerns.

Finally, AEOD concluded that the limited number of ESF actuations, the wide variety of ESF systems, and the differences in the types of ESF actuations make comparison between units very difficult. In those cases where frequent actuations are experienced at a unit, the information deriving from the events should be useful in appraising the performance of the licensees in resolving the problems on an individual basis.

Unplanned Reactor Scrams at U.S. Light Water Reactors

This study analyzed unplanned reactor scrams that occurred at U.S. nuclear power plants in 1984. The study is one of a series of periodic AEOD trends-and-patterns analysis reports which draw upon the more complete operational experience information now required for LERs. In this report, a reactor scram was defined as any unplanned actuation of the reactor protection system which resulted in control rod motion. The progression of events leading to reactor scrams and the post-scram response of the plant and personnel have obvious safety significance, and the Commission has concluded that a reduction in the frequency of challenges to plant safety systems should be a prime goal of each licensee.

Based upon the evaluations and analyses in this study, AEOD arrived at the following general observations with regard to reactor scrams:

- A reduction of hardware failures, primarily in balance of plant systems would significantly reduce the number of reactor scrams.
- There are a number of post-scram recovery complications which are due to equipment failures and personnel errors unrelated to the original scram cause that may have important safety implications.
- Approximately 50 percent of all reactor scrams caused by human error during plant operation at 15 percent power or more are traceable to activities by unlicensed personnel (instrument and control technicians, electricians, pipefitters, etc.).

In addition to these general observations, the report contains a number of specific conclusions based upon the analysis of the 494 reactor scrams wich were identified in 1984. Overall, AEOD observed a slight decline (9 percent) in the average rates of automatic and manual reactor scrams from 1983 to 1984, i.e., from 5.6 to 5.2 per plant per year, and from 0.9 to 0.7 per plant per year, respectively.

As part of this analysis, reactor scram rates for plants in other countries were also collected and analyzed. Only rough comparisons were possible because of the age and relative lack of documentation of foreign data. It appears, however, that the average reactor scram rates for the countries examined (France, Japan, West Germany, Sweden) were below those for the U.S. reactor population, both for boiling water and pressurized water reactors.

Table 3. Non-Reactor Engineering Evaluations

Designation	*Subject*	*Issued*
N501	Summary of the Non-Reactor Event Report Data Base for the Period January-June 1984	5/85
N502	Summary of the Non-Reactor Event Report Data Base for the Period July-December 1984	6/85
N503	Report of Medical Misadministrations for January 1984-December 1984	7/85

Safety Implications of In-Plant Pressurized Gas Storage and Distribution Systems at Nuclear Power Plants

This report, prepared for AEOD by the Nuclear Operations Analysis Center at Oak Ridge National Laboratory, addresses a study of the hazards associated with the use of compressed gases in nuclear power plants. Ten gases commonly used in the plants were selected to determine the safety implications associated with their storage and handling. The gases were air, acetylene, carbon dioxide, chlorine, Halon, hydrogen, nitrogen, oxygen, propane, and sulfur hexafluoride. The study was limited to the potential hazards from the time the gases were brought into the plant until they entered into the process, with special attention to the potential for subsequent events leading to secondary plant hazards.

The physical properties of the gases were reviewed, as were applicable industrial codes and standards. Incidents involving pressurized gases in general industrial use and in the nuclear industry were studied. In this report, general hazards such as missiles from ruptures, rocketing of portable cylinders, pipe whipping, asphyxiation, and toxicity are discussed. Even though some serious injuries and deaths over the years have occurred in industries handling and using pressurized gases, the study found that the industrial codes, standards, practices, and procedures are very comprehensive. It was recommended that the most important step to take in ensuring the safe handling of gases is to enforce these well known and established methods. Other recommendations for further improvement in the safe handling of pressurized gases were:

- Protection should be provided to safety-related equipment to prevent damage from gas cylinder missiles.
- Areas containing safety-related equipment should be protected from possible explosions resulting from rapid releases of hydrogen.
- Lines and tanks containing hazardous gases should be so designated by easily recognizable identification.

These recommendations are under review within the NRC to determine which additional actions by licensees should be required.

ANALYSIS OF NON-REACTOR OPERATIONAL EXPERIENCE

During fiscal year 1985, AEOD's Non-reactor Assessment Staff issued two semiannual summaries of the non-reactor event report data and the Report on Medical Misadministrations for calendar year 1984. The misadministration report covers diagnostic and therapeutic misadministrations, multiple misadministrations reported by licensees, and licensee proposed corrective actions. It states that during 1984, the number, types, and causes of diagnostic misadministrations were found to be about the same as reported for 1981 through 1983. However, while the numbers of therapy misadministrations reported for 1981 and 1984 are about the same (averaging 11), they are markedly higher than the number reported in 1982 and 1983 (four).

A preliminary case study on Therapy Misadministrations Reported to the NRC Pursuant to 10 CFR 35.42 was issued for peer review in June 1985. The regulation became effective in November 1980, as a result of numerous serious misadministrations in the 1970s. It requires the reporting by NRC licensees of diagnostic and therapy misadministrations involving nuclear medicine studies or radiation therapy. There are about 400 licensees authorized to perform teletherapy treatment, 600 authorized to perform brachytherapy treatment, and 600 authorized to perform adiopharmaceutical therapy treatment.

The Commission's purpose in requiring the submittal of misadministration reports to the NRC is to assure that their causes are properly identified and that licensees implement appropriate corrective actions to prevent recurrence. If potential generic problems are identified, the Commission notifies other licensees of the generic problem or concerns, and assesses the need for additional actions, e.g., changes in regulations to reduce the occurrence of similar and perhaps more serious events. This preliminary case study reviewed licensee reports submitted from November 1980 through July 1985, and will be issued as a final report in fiscal year 1986.

ABNORMAL OCCURRENCES

AEOD prepares the quarterly Report to Congress on Abnormal Occurrences (NUREG-0090 series) which feeds back significant event information to licensees, Government agencies, and the public. These reports are available from the GPO Sales Program, Superintendent of Documents, U. S. Government Printing Office, Post Office Box 37082, Washington, D.C. 20013-7982. (For a description of NRC's requirements under law to report abnormal occurrences, see the 1980 NRC Annual Report, p. 82). The reports issued during fiscal year 1985 were NUREG-0090, Vol. 7, No. 2 (April— June, 1984); Vol. 7, Vol. 3 (July-September 1984); Vol. 7, No. 4 (October-December 1984); and Vol. 8, No. 1 (January-March 1985). These reports covered 8 occurrences at nuclear power plants, 10 occurrences among fuel cycle facilities and other NRC licensees (industrial radiographers, medical institutions, industrial users, etc.), and 6 occurrences at Agreement State licensees. The reports also contained updated information for some abnormal occurrences which had been reported in previous fiscal years.

The abnormal occurrences reported during fiscal year 1985 are briefly described below. Additional details, as well as the updated information for some previously reported abnormal occurrences, are contained in the quarterly abnormal occurrence reports referenced above. Table 4 lists these reports and the occurrences covered in them.

Some of the events described below resulted in escalated enforcement actions, including civil penalties, by the NRC. (See Table 2 in Chapter 8 for listing of all civil penalties imposed during the report period, with capsule descriptions of the reasons therefor.)

Nuclear Power Reactors

Inoperable Containment Spray System. On March 17, 1984, Southern California Edison Company discovered that both of the containment spray pump manual discharge isolation valves at San Onofre Unit 3 (Cal.) were locked shut, rendering the independent containment spray systems inoperable. It was found further that the condition had existed for about 13 days, during which time the plant had operated at power levels up to full power. During this same period, another violation occurred which further degraded the containment heat removal system: for about one-and-a-half days, one of two diesel generators had been removed from service. This meant that if a total loss of off-site power had occurred, the emergency power source for the containment emergency fan cooler system would have been inoperable. Although there was no demand for the containment cooling systems to perform their accident-mitigating functions during the 13-day period, automatic actuation of the containment spray system would not have been possible had an actual loss-of-coolant accident occurred. The apparent underlying causes of these lapses were (1) inadequate review and approval of changes made to a previously established valve alignment check list, and (2) the existence of an administrative procedure, promulgated by management, which allowed such changes to be made without adequate review and approvals. The licensee revised the procedures and training programs in question.

On May 16, 1984, the NRC proposed imposition of a civil penalty in the amount of $250,000. The forwarding letter noted that other enforcement actions since January 1983 at the San Onofre Units 2 and 3 indicate that management problems have not been adequately corrected.

Based on the licensee's prompt and extensive corrective activity following the May 16, 1984 NRC letter, the civil penalty was reduced to $125,000 on September 24, 1984.

Degraded Isolation Valves in Emergency Core Cooling Systems. During the past several years, a number of events have occurred involving open valves—including check valves (valves designed to allow water to flow in one direction only)—in the emergency core cooling systems of various General Electric-designed boiling water reactors. Some of these events resulted in the high pressure reactor coolant system's overpressurizing either the low pressure coolant injection (LPCI) system (the low pressure mode of the residual heat removal, or RHR, system), the high pressure coolant injection (HPCI) system (the low pressure suction portion thereof), or the low pressure core spray system. All of these systems are designed to mitigate the consequences of a loss-of-coolant accident (LOCA). The events substantially reduced safety margins for preventing an interface LOCA. In some reactor designs, the possible interface LOCA could bypass the containment, with radioactive material being discharged to the outside environment. One event resulted in a partial draining of the reactor vessel. Most of the events were due to personnel errors. The events are briefly summarized below.

- On December 12, 1975, with the Vermont Yankee plant operating at 99 percent power, the utility, Vermont Yankee Nuclear Power Corporation, was performing monthly LPCI pump and motor-operated valve operability surveillance testing. Because of leaking valves, a flow path existed from the reactor vessel to the "A" LPCI loop, thereby pressurizing the loop in excess of its 450 psig system design pressure.

- On January 1, 1977, with the Cooper Nuclear station (Neb.) operating at about 97 percent power, the Nebraska Public Power District was performing a high pressure coolant injection (HPCI)—turbine trip and initiation logic functional test. The HPCI testable check valve failed to seat, allowing feedwater to flow backwards through the HPCI injection line into the HPCI suction piping.

- On September 14, 1983, while LaSalle Unit 1 (Ill.) was in cold shutdown, the Commonwealth Edison Company was performing a routine surveillance test of the RHR system relay logic. The injectable test valve stuck open, permitting between 5,000 to 10,000 gallons of water to drain from the reactor vessel. However, the reactor core remained covered at all times.

Table 4. Abnormal Occurrence Reports Issued During FY 1985

Occurrences at Nuclear Power Plants

Designation	*Subject*	***NUREG-0090 Issued***
84-6	Inoperable Containment Spray System	Vol. 7, No. 2 October 1984
84-8	Degraded Isolation Valves in Emergency Core Cooling Systems	Vol. 7, No. 3 April 1985
84-9	Degraded Shutdown Systems	
84-10	Loss of Off-site and On-site AC Electrical Power	
84-11	Refueling Cavity Water Seal Failure	
84-17	Four Control Rods Fail to Insert During Testing	Vol. 7, No. 4 May 1985
84-18	Degraded Upper Head Injection System Accumulator Isolation Valves	
85-1	Premature Criticality During Startup	Vol. 8, No. 1 August 1985

Occurrences at Fuel Cycle Facilities (Other than Nuclear Power Plants)

Designation	*Subject*	***Issued***
84-12	Degraded Material Access Area Barriers	Vol. 7, No. 3 April 1985
84-19	Buildup of Uranium in a Ventilation System	Vol. 7, No. 4 May 1985

Occurrences at Other NRC Licensees (Industrial Radiographers, Medical Institutions, etc.)

Designation	*Subject*	***Issued***
84-7	Therapeutic Medical Misadministration	Vo. 7, No. 2 October 1984
84-13	Contaminated Radiopharmaceuticals Used In Diagnostic Administrations	Vol. 7, No. 3 April 1985
84-14	Therapeutic Medical Misadministration	
84-15	Significant Internal Exposure to Iodine-125	
84-16	Therapeutic Medical Misadministration	
85-2	Diagnostic Medical Misadministration	Vol. 8, No.1 August 1985
85-3	Diagnostic Medical Misadministration	
85-4	Unlawful Possession of Radioactive Material	

Occurrences at Agreement State Licensees

Designation	*Subject*	*Issued*
AS84-2	Contaminated Radiopharmaceuticals Used in Diagnostic Administrations	Vol. 7, No. 3 April 1985
AS84-3	Overexposure of a Radiographer Trainee	Vol 7, No. 4 May 1985

Occurrences at Agreement State Licensees)

Designation	*Subject*	*Issued*
AS85-1	Overexposure of an Employee	Vol. 8, No.1 August 1985
AS85-2	Radiation Hand Burn to an Assistant Radiographer	
AS85-3	Overexposure of an Assistant Radiographer	
AS85-4	Lost Well Logging Source	

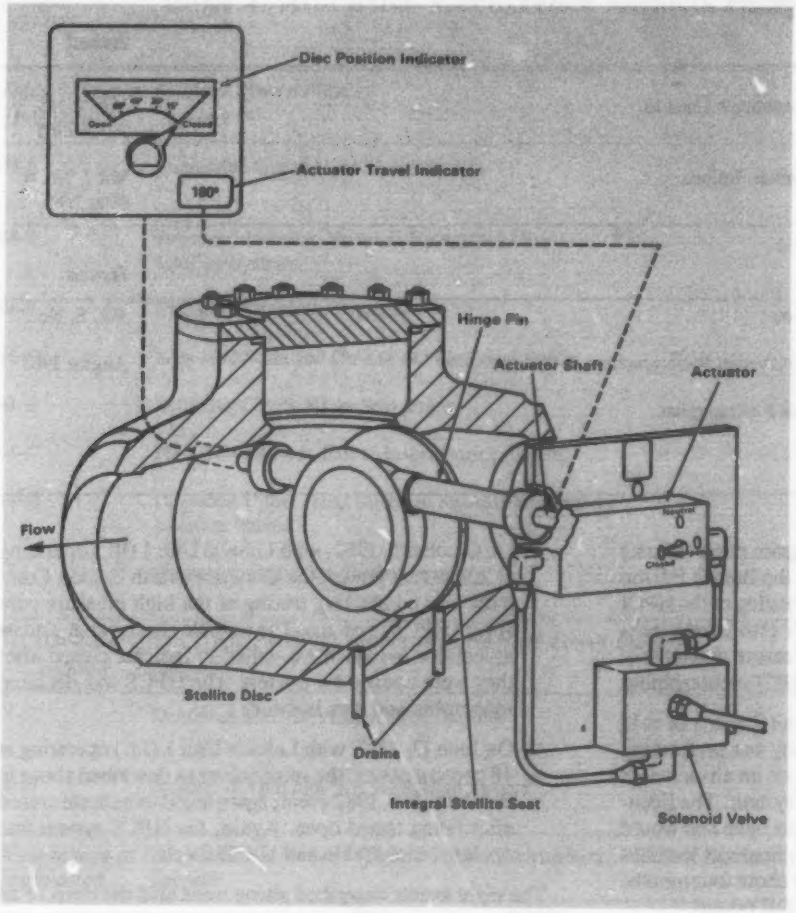
- On September 29, 1983, with the Pilgrim plant (Mass.) operating at about 96 percent power, the Boston Edison Company was performing functional testing of the HPCI system logic. As a result of an operator error, and a leaking testable check valve, feedwater pressure overpressurized the low pressure portion of the HPCI suction piping.
- On October 28, 1983, with Hatch Unit 2 (Ga.) in cold shutdown, the Georgia Power Company was performing a valve operability testing procedure on an air-actuated check valve on one train of the RHR system. The licensee discovered that the check valve was open and would not close. The condition, caused by incorrect installation of air supply lines, had existed for about four months while the plant operated at close to full power.
- On August 14, 1984, with Browns Ferry Unit 1 (Ala.) operating at about 100 percent power, the Tennessee Valley Authority was performing a core spray (CS) logic functional test. Because of operator error, and also improper maintenance on a testable check valve (which resulted in the valve's being open while indicating closed), the high pressure reactor coolant system (about 1,050 psi) was open directly to the low pressure CS system (designed for 500 psi).

The above six events were reported in Abnormal Occurrence Quarterly Report, NUREG-0090, Vol. 7, No. 3, for July-September 1984. In addition, two similar events were discovered in data searches and will be included in an update early in fiscal year 1986 (Vol. 8, No. 2). These events are briefly described below.

- On October 5, 1982, with LaSalle Unit 1 (Ill.) operating at 20 percent power, the Commonwealth Edison Company was conducting testing of the high pressure core spray (HPCS) systems. The testable check valve, and its associated bypass valve, failed to indicate closed after they were opened for the test. The HPCS was declared inoperable and was isolated.
- On June 17, 1983, with LaSalle Unit 1 (Ill.) operating at 48 percent power, the same valves as described above in the October 5, 1982 event, again failed to indicate closed after being tested open. Again, the HPCS system was declared inoperable and was isolated.

The eight events described above were also the basis of an AEOD case study (C501) which was issued in September 1985. (See also "Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors," under "Analysis of Reactor Operational Experience," earlier in this chapter.)

Degraded Shutdown Systems. On June 23, 1984, Public Service Company of Colorado discovered that six out of the total 37 control rod pairs at the Fort St. Vrain facility failed to insert upon receipt of an automatic scram signal from the plant protection system. At the time of the event, the plant was operating at 23 percent power. A somewhat similar event had occurred previously on February 22, 1982, when two control rod pairs failed to insert automatically during a manual scram (i.e., a scram initiated by the operator); however, for this event, the reactor was already in a subcritical condition during routine startup operation.



Check valves, such as the one shown here, are designed to ensure that the flow of water can be in one direction only; they are important components in the emergency core cooling system of General Electric boiling water reactors. In the past several years, a number of unplanned events have resulted from faulty installation, maintenance or operation of such valves which substantially reduced safety margins in the capacity to prevent more severe accidents in the plants involved. A summary of the events is given in the text.

Subsequently, there were additional problems at this plant. During July 1984, the licensee reported that numerous control rod position instrumentation anomalies had occurred. Then, on November 5, 1984, with the plant still shut down since the June 23, 1984, event, a portion of the plant's redundant rapid shutdown systems was tested and failed to operate properly; many of the borated graphite balls failed to drop from a hopper in the reserve shutdown system.

On its own merit, the June 23, 1984 event is a safety concern. The failure of six control rod pairs to automatically insert upon receipt of a valid scram demand signal is a common-mode failure that constitutes a partial anticipated-transient-without-scram (ATWS). When viewed in the context of the other related occurrences, the event takes on additional safety significance.

The NRC conducted an assessment of the licensee's overall operation. The assessment team found significant weakness in every area of the operation that was audited. The report on the assessment included both short term and long term recommendations. The overall conclusion of the assessment team was that the Fort St. Vrain facility should not be allowed to restart until all weaknesses had been addressed.

The licensee refurbished all control rod drive mechanisms, repaired the control rod position instrumentation, replaced all boron carbide balls with new material in the reserve shutdown systems, and took other appropriate actions to resolve the concerns in the special NRC assessment report. On July 19, 1985, the licensee was authorized to resume power operations up to 15 percent of rated thermal power.

Loss of Off-site and On-site AC Electrical Power. On July 26, 1984, Pennsylvania Power and Light Company was conducting startup testing (a loss of off-site power test) on Susquehanna Steam Electric Station Unit 2 with the unit operating at 30 percent power. An operator mistakenly opened the wrong switch to each of four 4160V engineered safety system busses; as a result, the emergency diesel generators were unable to automatically start as required upon loss of off-site power. Recovery from the event was complicated since most instrumentation in the control room failed downscale.

There was no direct impact on public health or safety from the event. Even though some safety-related equipment was disabled—equipment designed to mitigate the consequences of design basis accidents, in the unlikely event that one occurred—the high pressure coolant injection system and the reactor core isolation cooling systems were available to provide makeup water to protect the core until power was restored.

The causes of this event were identified as inadequate implementation of corrective action for previously identified problems, inadequate human engineering of the local control panels, ineffective independent verification, imprecise procedures, inadequate operator training, and operator error.

Immediately after the incident, the licensee initiated an investigation and instituted immediate and long term corrective actions. The immediate corrective actions were completed prior to NRC permission to restart the plant. The licensee plans to identify to the NRC the status and/or schedule for completing the long term items. The licensee assessed the event's impact and lessons learned on Unit 2, and applied the appropriate immediate and long term corrective actions to Unit 1 as well.

On December 18, 1984, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of a Civil Penalty in the amount of $50,000.

Refueling Cavity Water Seal Failure. During a refueling outage on August 21, 1984, Connecticut Yankee Atomic Power Company notified the NRC that the reactor refueling cavity water seal had failed at the Haddam Neck Plant, draining the refueling pool water to the containment floor. When the seal assembly failed, water was dumped around the neutron shield tank surrounding the reactor vessel and into the containment sump below the vessel. The sump overflowed into the containment floor drains and onto the lower level of the containment. Water also leaked out around the reactor coolant loop penetration piping and wetted components inside the loop areas of containment.

The licensee restricted containment access and replaced the reactor vessel head to minimize radiation exposure and to protect the reactor core internals. The licensee suspended refueling operations until a failure analysis and corrective actions were completed and until the NRC reviewed and approved the plant recovery program.

There were serious potential safety consequences which could have resulted not only from this refueling incident but also could have occurred during previous refueling operations when the same design of seal was used. For example, the spent fuel pool (SFP) gates would have been opened within an hour of the seal design failure, and handling of spent fuel assemblies in the refueling cavity could have been in progress within 18 hours. If refueling had been in progress, as many as four spent fuel assemblies could have been partially or fully uncovered as the reactor cavity drained. In addition, the top three feet of all fuel assemblies in the SFP would have been uncovered if the pool had drained through the transfer tube. Fuel assemblies recently removed from the reactor vessel would be even more radioactive (and generate considerably more decay heat) than spent elements stored for some time in the SFP. In all cases, there was the potential for fuel rod damage from overheating, with subsequent release of gaseous fission products from the damaged fuel rods. In addition, loss of the water would reduce the radiation shielding for spent fuel. This would have increased the radiation field in the refueling areas which could have precluded those operator actions necessary to prevent overheating of fuel being moved to or stored in the SFP.

The event was caused by inadequate design of the bladders used in the seal assembly. The bladders were neither specified nor suitably tested to withstand, with a suitable safety margin, the hydrostatic pressure expected to occur during normal use. Post-event inspection of the seal assembly revealed that the outer bladder had extruded between the steel plates for about one-quarter of the seal circumference. After subsequent testing, the licensee concluded that there was not sufficient margin in the seal design to prevent extrusion of the bladder through the two-inch gap. The design verification by an independent engineer, the safety evaluation, and the review by the on-site and off-site-review committees all failed to discover the inadequate seal design.

The licensee immediately initiated a recovery program which included containment dewatering/decontamination; equipment damage assessment; seal failure analysis and seal modifications; and procedure review.

On December 13, 1984, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of a Civil Penalty in the amount of $80,000; in addition, an Order modifying the license was imposed requiring review and appraisal by an independent organization of (1) design modification packages approved since January 1, 1979, in order to determine the adequacy of design control and to determine whether each such modification introduced any previously unanalyzed failure mode or mechanism; and (2) the process for initiating, evaluating, reviewing, approving, and implementing design change modifications, in order to determine if deficiencies exist in the process, and to provide recommendations for improvement.

Four Control Rods Fail to Insert During Testing. On October 6, 1984, Pennsylvania Power and Light Company was performing quarterly individual control rod scram testing on Susquehanna Unit 1 while the plant was operating at 60 percent power. Because of a common mode failure of the scram pilot solenoid valves (SPSVs), four rods failed to insert and nine others stalled before scramming.

The safety significance of the event was the reduction in the required "extremely high probability" of shutting down the reactor in the event of an anticipated operational occurrence. The reduction consisted in the following: (1) following the common mode failure of the SPSVs, there was a potential for a significant number of control rods to be inoperable (the mechanism that could have possibly identified the problem earlier, the surveillance procedure, was not properly reviewed, and therefore a precursor event on June 13, 1985 was not recognized and investigated); and (2) even though the plant has backup scram valves, at the time of the event the condition of the valves was not known, because they had not been tested since before the plant originally started up.

The SPSVs failed when the disc holder subassembly disc stuck to the seat on the T-ASCO valve bodies. The cause of the failure was believed to be contamination of the polyurethane seat material by oil and/or water which had been introduced into the control rod drive instrument air system. The licensee continued its investigation to determine the exact nature and source/origin of the contaminants found in the instrument air system. The seat material was replaced by Viton-A, which is resistant to all oils that could be introduced into the instrument air system as well as to water and other chemical contaminanants.

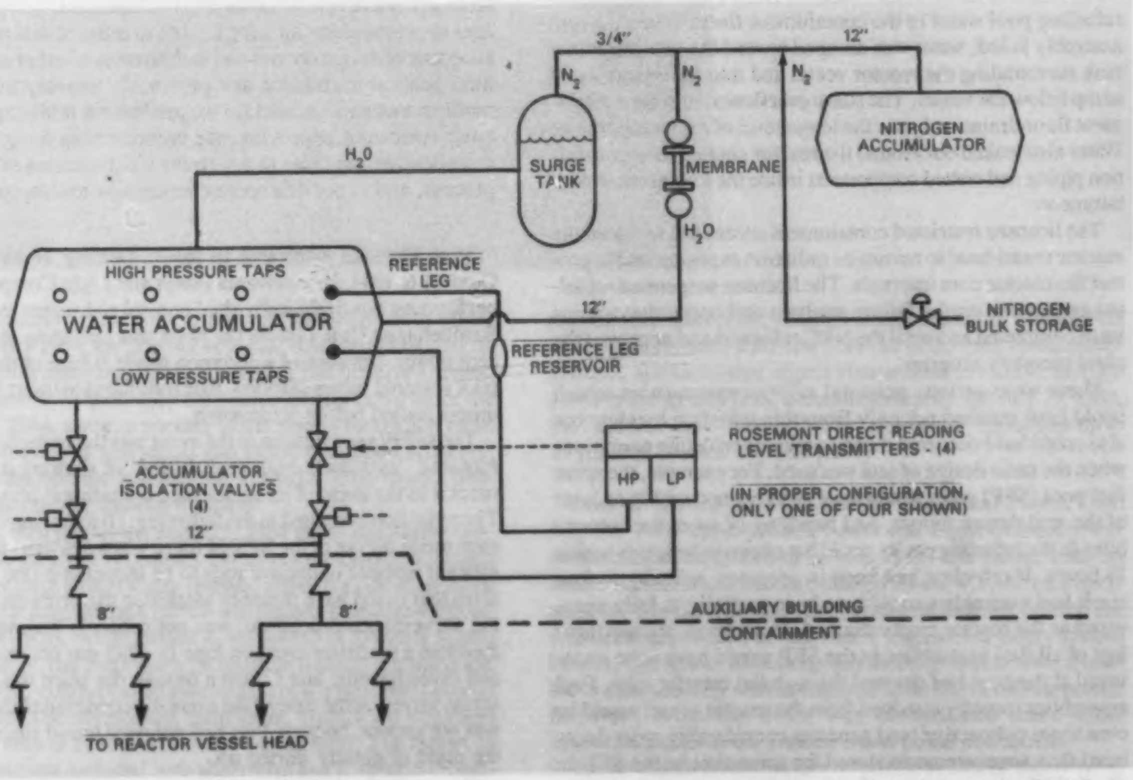
A contributing cause was the licensee's inadeqate review of the data associated with a full core scram which occurred on June 13, 1985. The data provided information by which the deficiency might have been identified, before some rods actually failed to insert.

Degraded Upper Head Injection System Accumulator Isolation Valves. On November 1, 1984, the upper head injection (UHI) system accumulator isolation valves were discovered to be incapable of the required automatic closure for Duke Power Company's McGuire Nuclear Station, Unit 1 (N.C.). The UHI system is an engineered safety feature, designed to provide cooling of the core (by means of borated water) during the blowdown portion of the postulated loss-of-coolant accident (LOCA) transient for a large rupture in the cold leg of the reactor coolant system (RCS).

On November 1, 1984, while the licensee was draining the tank, it was discovered that the four isolation valves had failed to close on accumulator water low level. Investigation showed that the valves had been incapable of the required automatic closure since April 25, 1984. From that date until the condition was discovered on November 1, 1984, the plant had been operating for about five months. During that period, if there had been a large LOCA, a considerable amount of nitrogen could have been injected into the reactor vessel upper head. Although the effects of injecting this non-condensable gas has not been analyzed in detail, it is possible that it could interfere with cooling of the reactor core during such an accident. This condition is beyond the design bases for the plant and is not specifically analyzed in the safety analysis report.

Investigations revealed that the water accumulator differential pressure transmitters—which sense accumulator water level and provide the initiating signal for isolation valve closure—had been improperly installed on Unit 1. The impulse lines were not connected to the appropriate transmitter ports.

A new installation procedure was issued which requires verification of proper tubing connections for differential pressure transmitters; the licensee also committed to strengthening the post modification testing program.


3/4"
12"
N_2
N_2
N_2
NITROGEN ACCUMULATOR
H_2O
SURGE TANK
MEMBRANE
H_2O
HIGH PRESSURE TAPS
REFERENCE LEG
WATER ACCUMULATOR
12"
NITROGEN BULK STORAGE
LOW PRESSURE TAPS
REFERENCE LEG RESERVOIR
ROSEMONT DIRECT READING LEVEL TRANSMITTERS - (4)
HP
LP
ACCUMULATOR ISOLATION VALVES (4)
(IN PROPER CONFIGURATION, ONLY ONE OF FOUR SHOWN)
12"
8"
8"
AUXILIARY BUILDING
CONTAINMENT
TO REACTOR VESSEL HEAD

Between April and November of 1984, Unit 1 of Duke Power Company's Maguire Nuclear Station in South Carolina operated for about five months with four critical accumulator isolation valves lodged in the open position. NRC analysis showed that, had a loss-of-coolant accident occurred, nitrogen entering the upper head of the reactor vessel might have interfered with the emergency cooling process. Shown here is a simplified diagram of the Maguire unit's upper head injection system's piping and instrumentation. Failure of the isolation valves to close resulted from improper installation of pressure transmitters.

On February 20, 1985, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $50,000. The NRC Office of Nuclear Reactor Regulation (NRR) undertook to assess the effects of the accumulator isolation valves failing to close during a large LOCA in a plant with a UHI system. NRR is also considering initiation of additional studies regarding the net safety benefit of the UHI system and changes in the technical specification requirements. The licensee is also investigating the efficacy of removal of the UHI system.

Premature Criticality During Startup. On February 28, 1985, while South Carolina Electric and Gas Company was starting up the Virgil C. Summer Nuclear Station Unit 1, an unanticipated transient occurrence resulted in a high flux positive rate trip (automatic plant shutdown).

During a nuclear power plant startup, control rods are withdrawn in a predetermined sequence to achieve criticality. In order to avoid rapid increases in power, three barriers of defense are used, namely, personnel performance, procedural control, and reactor instrumentation to automatically scram (trip) the reactor. For this event, the first two barriers failed. Consequently, the advent of criticality was not recognized and rod withdrawal was continued until the startup rate approached rapidly (by later estimates, 16 to 17 decades per minute (dpm)).

The actual safety consequences of the event were minimal, but the event is significant because it represented an unnecessary challenge to the reactor protection system, and also because the reactor was not being properly controlled during plant startup.

The cause was primarily the failure of the shift supervisor (who was responsible for the actions of the trainee who was pulling rods) to be fully aware of plant status, to colsely monitor instrumentation, and to anticipate criticality whenever rods are being withdrawn, as required by station procedures.

Contributing to, but not justifying, this failure to monitor and anticipate criticality was a calculated estimated critical position which was in error by more than 125 rod steps. The error in estimated critical position resulted, primarily, from procedural inadequacies.

The shift supervisor was removed from duty until the licensee completed an evaluation of the event, its causes, and the supervisor's capability to continue licensed operator duties. The supervisor was given formal counseling for failure to maintain an awareness of plant conditions during reactor startup. The supervisor resumed licensed operator duties on March 13, 1985. Procedures were also revised.

The NRC issued a notice of violations to the licensee on April 3, 1985.

Fuel Cycle Facilities and Other NRC Licensees

Therapeutic Medical Misadministration. Between April 11, 1984, a patient at the St. John's Medical Center, Anderson, Ind., received a 3,200 rad therapautic radiation exposure to the rear chest wall, instead of the intended 2,000 rad exposure. The cause was an error in the computer calculations in the treatment plan for the patient.

There were no significant consequences detected at the time, but, because of the exposure, the patient was subject to an increased risk of radiation pneumonitis for the remainder of the right lung.

Degraded Material Access Area Barriers. On May 22, 1984, an NRC Physical Security Inspector conducting a routine physical security inspection of the Nuclear Fuel Services facility in Erwin, Tenn., discovered three areas of degradation in a material access area (MAA) boundary which lessened the overall effectiveness of the security barrier. A subsequent survey by the licensee of all MAA barriers identified four additional boundary degradations.

The barrier degradations occurred as a result of inadequate communication between the facility maintenance and security personnel.

In addition, the licensee's management and administrative control systems failed to promptly detect and correct the MAA boundary degradations.

On July 27, 1984, the NRC proposed a civil penalty of $100,000. However, because of the licensee's prompt and extensive corrective action, the NRC mitigated the penalty to $80,000, on January 22, 1985.

Contaminated Radiopharmaceuticals Used in Diagnostic Administrations. On May 18, 1984, two nuclear pharmacies (Nuclear Pharmacy, Inc., located at Chicago, Ill., and Syncor International, Inc., located at Blue Ash, Ohio) received faulty devices from the Medi-Physics, Inc., facility located at Tuxedo, N. Y., which were used for preparing doses of technetium-99m, a radiopharmaceutical widely used for diagnostic medical tests. The radiopharmaceutical was contaminated with molybdenum-99, another radioactive isotope.

Contrary to NRC license conditions, the contaminated radiopharaceuticals were shipped by the nuclear pharmacies to hospitals, resulting in 28 patients' receiving unnecessary exposures during diagnostic medical tests.

Based on NRC inspections, the causes of the violations of the licensees were as follows:

(1) At Syncor, the responsible radiopharmacist apparently at first misinterpreted the test results. However, having become aware of the contaminated technetium-99m, he apparently made no attempt to notify clients or recover the technetium shipments he knew were contaminated.

(2) At Nuclear Pharmacy, no breakthrough tests were performed. In addition, until the NRC became involved, the licensee—although aware that a breakthrough had occurred—did not take action to notify its clients and to determine whether contaminated material had actually been used on patients. Even after the licensee began investigating the matter, the licensee did not provide the NRC with reliable information.

Both licensees have upgraded their procedures to assure that the NRC-required molybdenum-99 breakthrough tests are completed and fully evaluated before the technetium-99m is distributed to customers.

On January 2, 1985, the NRC issued to Syncor an Order Imposing a Civil Monetary Penalty in the amount of $8,500. On October 26, 1984, the NRC forwarded to Nuclear Pharmacy an Order Modifying Licenses, Effective Immediately. The Order included specific changes in the licensee's procedures and weekly audits by each facility manager of all NRC-licensed activities. In addition, the licensee was directed to obtain the services of one or more qualified independent organizations to assess the licensee's qualifications and competence to operate.

Therapeutic Medical Misadministration. On July 3, 1984, Washington University Medical Center, St. Louis, Mo., reported to the NRC that a 64-year old patient had received a series of radiation treatments totaling 6,400 rads of radiation, instead of the prescribed dose of 4,000 rads. The cause was an error in the treatment plan.

The licensee hospital subsequently reported that the patient had had no acute, serious side-effects as a result of the higher radiation doses. An NRC medical consultant, retained to evaluate this case, stated that the total radiation dose to the head was still within the acceptable range for such treatments.

Significant Internal Exposure to Iodine-125. On August 8, 1984, the NRC Region I Office was notified by the Veterans Administration Medical Center, Bronx, N. Y., that on August 3, 1984, an individual (Individual A) working at the licensee's facility had been found to have a thyroid burden of approximately 524 microcuries of iodine-125 (apparently received on July 28, 1984), an amount which greatly exceeded the allowable NRC limits. Three co-workers (Individuals B, C, and D) also showed thyroid burdens, but at values within regulatory limits.

The licensee projected a total absorbed dose to Individual A's thyroid of approximately 2,000 rads, based on maximum measured uptake of 524 microcuries. The licensee's consulting physician does not anticipate significant thyroid damage to occur, although some loss of thyroid function is possible. The thyroid burdens for Individuals B, C, and D were approximately 156, 94, and 68 nanocuries, respectively. These would result in small thyroid exposures, well within NRC regulatory limits.

The most likely cause of the event appeared to be pipetting by mouth or other poor laboratory practice in the handling of iodine-125.

To empnasize the importance of adherence to NRC requirements and safe performance of licensed activities, an Order Modifying License was issued by the NRC to require periodic unannounced audits of the licensee's radiation safety program by an independent third party.

Therapeutic Medical Misadministration. On August 15, 1984, the NRC was notified by the United States Air Force Medical Center at Keesler Air Force Base near Gulf Port, Miss., of a therapeutic medical misadministration which had been discovered on August 9, 1984. A patient received a total exposure to a large portion of the left lung of 2,475 rads, instead of the intended 1,500 to 1,800 rads. No immediate adverse health effects were detected as a result of the overexposure; however, the licensee agreed that the risk of radiation pneumonitis and radiation-induced fibrosis were significantly increased as a result of this event.

The overexposure occurred because the administrative procedures for control of the treatment were inadequate.

Buildup of Uranium in a Ventilation System. On October 5, 1984, Nuclear Fuel Services, Inc., in Erwin, Tenn., notified the NRC that an excessive buildup of uranium had been discovered in the new ventilation system (including a scrubber) of the scrap recovery facility at their plant.

Even though it was determined that a criticality event could not have occurred, the event was significant in that the accumulation of uranium-235, in the scrubber and ducting, was considerably greater than the usual, assuredly safe, level. The primary cause of the uranium buildup was equipment design. A contributing cause was the licensee's failure to take appropriate corrective actions when material control action limits were exceeded.

On February 21, 1985, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $20,000. In addition to the civil penalty, the NRC letter also enclosed an Order Modifying License. The Order amends the license to require the licensee to expand the duties and responsibilities of its Internally Authorized Change Council.

Because of extensive and comprehensive corrective action taken by the licensee, the NRC reduced the civil penalty to $15,000, on November 27, 1985.

Diagnostic Medical Misadministration. On January 7, 1985, a representative of St. Luke's Hospitals, Chesterfield, Mo., reported to the NRC Region III that, on December 19, 1984, a female patient received a radioactive material other than that prescribed for a scheduled diagnostic medical test. As a result, the patient's thyroid received a therapeutic dose in the range of 6,500 to 9,000 rads, rather than the expected dose of about 200 millirads.

The patient did not exhibit any apparent immediate injury. However, the patient has a possibly 50 percent chance of developing hypothyroidism in the future. This possible decrease in thyroid function is not considered life threatening.

The misadministration was caused by the technologist's dispensing the iodine-131 capsule without a written requisition specifying the drug or the dosage, an action contrary to the Department's rules.

Diagnostic Medical Misadministration. On March 11, 1985, a representative of Tolfree Memorial Hospital of West Branch, Mich., reported that a patient had received a diagnostic radiation exposure on March 7, 1985, that was approximately 10 times the intended exposure. Because of inadequate procedures, the patient was administered 1,000 microcuries of iodine-131 instead of the normal amount of 100 microcuries. However, the licensee reported that no biological harm was expected because of the patient's age (70 years). The percentage of iodine-131 uptake generally decreases with age.

Unlawful Possession of Radioactive Material. On March 26, 1985, John C. Haynes, doing business as the John C. Haynes Company ("the licensee"), Newark, Ohio, was arrested by agents of the Federal Bureau of Investigation on charges of illegal possession and use of radioactive material (americium-241), and for making false statements to the NRC. On April 5, 1985, the NRC issued an Order requiring him to provide access to his laboratory facility for cleanup and removal of radioactively contaminated equipment. Following the cleanup, the license was to be revoked.

Americium-241 is hazardous in powder form if it becomes airborne and is inhaled by an individual. Lodged in the body, the americium may, over a long period of time, cause cancer. The principal hazard outside the laboratory would be the spread of americium through fire, vandalism, or other means. Such a dispersion of americium could represent a significant health hazard outside the house, but the hazard would diminish substantially with distance.

Agreement State Licenses

Contaminated Radiopharmaceuticals Used in Diagnostic Administrations. On January 30 and 31, 1984, approximately 16 patients at Rhode Island Hospital, located in Providence were administered technetium-99m which was contaminated with molybdenum-99, another radioactive material. Consequently, these patients received unnecessary exposures during diagnostic medical tests.

The licensee later made estimates of the dose to the liver from the molybdenum-99 for the 16 patients. The estimates ranged from about 20 rads to 120 rads, rather than the expected few rads.

The principal cause of the incident was the failure of a staff technologist to properly perform the molybdenum-99 breakthrough assay prior to injecting a technetium-99m based radiopharmaceutical.

Overexposure of a Radiographer Trainee. On October 31, 1984, a radiographer trainee, employed by Ultrasonics Specialists, Inc., of Amelia, La., received a significant exposure of 2,500-3,000 rads to his right index finger from an unshielded source, while working at Avondale Shipyards in Morgan City, La. The exposure resulted in a radiation injury.

The two principal factors contributing to the overexposure were (1) the trainee was performing industrial radiography without the direct supervision of an authorized instructor; and (2) a survey meter, which would have indicated that the source was not in the shielded position, was not used.

Overexposure of an Employee. Between June 1, 1984, and June 4, 1984, an individual employed by Gulf Nuclear, Inc. in Webster, Tex., received a whole body exposure of 29.2 rems while disassembling a radiographic exposure device. A physical examination and laboratory findings from medical tests performed on the technician showed no detectable symptoms of radiation exposure.

The apparent cause of the overexposure was that a source tag, indicating that a source was in the camera, was removed before the source was removed, and the employee assumed there was no source in the device. A secondary cause appears to be a lack of the training that would enable the employee to realize when a source was present and take appropriate action. It also appears that, at the time of the incident, management was not active to the point of providing sufficient supervision over handling procedures.

Radiation Hand Burn to an Assistant Radiographer. On August 24, 1984, the Texas authorities were notified by Baytown Industrial X-Ray of Houston that they had interviewed a radiographer who had a suspicious looking wound on his right hand. On August 27, 1984, the radiographer called the State agency and indicated that the wound had been diagnosed as a radiation burn. The incident occurred while the individual was working as an assistant radiographer for QA Special Services of Houston. The incident was attributed to exposure to an unshielded source. The individual may have received an estimated 2,000 rems to the palm of his hand when pushing it against the camera.

The immediate cause of the incident was failure to retract the source to its fully shielded position. The root cause of the incident appears to be poor training of employees by QA Special Services.

Overexposure of an Assistant Radiographer. Magnaflux Industrial Radiography Company of Houston, Tex., reported to the State agency that on November 19, 1984, an assistant radiographer received an overexposure from a radioactive source. It was later estimated that the employee had received about 1,320 rems (beta and gamma) to his right hand. The overexposure was due to an unshielded source.

The overexposure occurred for several reasons. First, the survey meter used by the assistant radiographer was found to have a crack in the anode, which caused the Geiger-Mueller tube to short out. The survey meter would function properly until it was put into a radiation field of 300-400 mR/hr. The readings would then become erratic and at times would fall to zero. In addition, someone had disabled the area monitor in the vault. Another contributing factor was that the assistant radiographer left the source unattended and, upon returning, assumed the source had been secured by someone else.

Lost Well Logging Source. On February 13, 1985, Schlumberger Well Service of Houston discovered that a 1.5 curie cesium-137 source was missing from its shield. The source was to be used by a crew from the licensee's Graham, Tex., facility.

On April 12, 1985, the licensee reported that the source had been found in a cow pasture approximately 120 feet north of a farm-to-market road three miles from the town of Graham, Texas. The serial number verified that this was the missing source. Since the source was not found near any of the routes taken by the licensee's trucks and was too far from the road to account for accidental loss, the licensee reported that the source had been stolen.

Nuclear Materials

CHAPTER 5

The NRC's Office of Nuclear Material Safety and Safeguards (NMSS) administers the regulation of nuclear materials, as distinct from nuclear reactor facilities (discussed in Chapters 2 and 3). NMSS conducts this regulation under three broad programs: fuel cycle and material safety, discussed in this chapter; materials and facilities safeguards, discussed in Chapter 6; and waste management activities, discussed in Chapter 7.

Activities discussed in this chapter include licensing and other regulatory activities concerned with (1) conversion of uranium ore concentrates (after mining and milling) to uranium hexafluoride; (2) conversion of enriched uranium hexafluoride to ceramic uranium dioxide pellets and subsequent fabrication into light water reactor fuel; (3) production of naval reactor fuel; (4) storage of spent reactor fuel; (5) transportation of nuclear materials; and (6) production and use of reactor-produced radioisotopes ("byproduct material").

Highlights of actions taken during fiscal year 1985 include:

- Completion of 45 major and 46 minor licensing activities dealing with fuel cycle plants and facilities.
- Completion of 143 design certification reviews for transportation packages.
- Completion of nearly 6,000 licensing actions on applications for new byproduct materials licenses and amendments and renewals of existing licenses. Nearly 5,300 of these actions were completed by the five Regional Offices; the remainder were completed at Headquarters.
- Transfer of most Federal agency materials licensing from NMSS Headquarters to the Regions in April 1985.

FUEL CYCLE ACTIONS

Licensing Actions

Licensing actions associated with the possession and use of source and special nuclear material continue to consume a large portion of staff effort. Special nuclear material licenses were issued at reactor sites to allow early receipt and storage of fuel prior to receipt of the Operating License. A major amendment related to expanded uranium recovery operations and improved waste treatment was completed and issued to the General Electric Corporation for their Wilmington, N.C., facility. The Westinghouse Electric license for the Columbia, S.C., facility was renewed, and the renewal included approval to use a new dry process for the conversion of enriched uranium hexafluoride to reactor fuel.

Kerr-McGee has applied for permission to operate a UF_6 to UF_4 conversion process at the Sequoyah UF6 production plant. The request was under staff review at the close of the report period. In response to a request for a hearing concerning the new activity, the Commission granted the request and directed that an informal hearing be held. Parties in the hearing will be the intervenors and Kerr-McGee; the staff will not be involved as a party. The proceeding was to be held beginning in late 1985.

Babcock & Wilcox (B&W) has converted a portion of their former plutonium plant at Parks Township, Pa., for use as a repair and refurbishment of reactor components. In addition, B&W has requested an amendment to authorize the use of an incinerator and high-force compactor at the Parks Township plant. These two components comprise a Volume Reduction Service Facility. The request was under staff review at the close of the report period. An informal hearing on this amendment request has been instituted in response to petitions from members of the public. The proceeding, which has begun, should be completed in 1986. The staff is not a party to the proceeding.

Decommissioning and Decontamination

Decommissioning and decontamination activities are underway at five former plutonium fuel facilities, four uranium fuel fabrication plants and a number of facilities possessing source material. Four plutonium facilities are presently in the process of decontamination, and a fifth facility has essentially been decontaminated. Two additional plutonium facilities have been decontaminated and are now being used for other nuclear-related purposes. Of the uranium fuel fabrication facilities, the decontamination of two plants is essentially complete. Licenses for the plants where decontamination is complete will be terminated, or modified if other nuclear activities are planned there. At the remaining plants, equipment is being removed and shipped to disposal sites.

Decommissioning and decontamination activities are also under way at a number of source material facilities. At the end of fiscal year 1985, there were approximately six licensed facilities undergoing decontamination, with several actions nearing completion. At several sites, decommissioning awaits a decision on where low-level waste, such as process slags, may be disposed.

Kerr-McGee. In the hearings involving the decommissioning of the Kerr-McGee Rare Earths Plant in West Chicago, Ill. (see 1984 NRC Annual Report, pp. 69, 70.), and the stabilization of the plant wastes, the Hearing Board directed the staff to supplement the Environmental Statement to determine, among other things, the acceptability of Kerr-McGee's application for permanent disposal of nuclear wastes on-site under the Uranium Mill Tailings Radiation Control Act of 1978, as amended, (UMTRCA), and to review the application pursuant to the National Environmental Policy Act. Criteria had not been established for disposal under UMTRCA at the time the Environmental Statement was published; therefore, the statement evaluated storage of waste, but not disposal. The staff estimates that the final supplement to the Environmental Statement will be completed in March 1987. The hearing will take place after the supplement is issued.

In the proceeding involving the NRC Order to Kerr-McGee related to the cleanup of Kress Creek (near the West Chicago site), the Hearing Board, in an Order issued on September 26, 1985, scheduled the hearing to begin on January 6, 1986, and to continue until concluded.

DOE "UMTRCA" Site. The staff has continued to work with the Department of Energy (DOE) on the remedial action required under Title I of the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978, for the Canonsburg, Pa., site. On-site remedial action is scheduled to be completed in November 1985. Once remedial action is complete, the NRC will issue a license to DOE, as site owner, for the care, maintenance, and monitoring of the radioactive material stabilized at the site.

Special Sites. Under the "Special Sites" Section (Section 151(c)) of the Nuclear Waste Policy Act of 1982, title to low-level waste generated as a result of recovering zirconium, hafnium, or rare earths from source material, and also the land upon which the wastes are disposed, shall be transferred to DOE upon request of the owner. However, such transfer can occur only after the site has benn decontaminated and stabilized in accordance with NRC requirements, and after the owner has made adequate financial arrangements, approved by NRC, for long-term maintenance and monitoring.

West Valley Demonstration Project. The Commission continued its safety oversight role for the West Valley (N.Y.) Demonstration project in 1985. The Department of Energy (DOE) completed the Project Plan, as required by the West Valley Demonstration Project (WVDPA), and received Commission comments. The schedule shown in the Plan indicates that the vitrification process for high-level waste may be started in 1988 in West Valley. Safety Analysis Report modules are being prepared by DOE for various aspects of the Project. The Commission is reviewing these reports and evaluating public health and safety aspects of the Project.

The Commission completed a report and additional studies on the Facility Disposal Area (FDA) currently being used at the West Valley site. The report provides information on the confinement capability of the FDA.

In accordance with the West Valley Demonstration Project Act, the Commission has monitored the DOE contractor's quality assurance and environmental surveillance programs. Suggestions for improvements to these programs were provided.

Interim Spent Fuel Storage

The Nuclear Waste Policy Act of 1982 (NWPA) clearly established that utilities have the primary responsibility for interim storage of their spent fuel until a Federal repository or monitored retrievable storage (MRS) installation, projected for 1998, is available. Although some contingency storage is available from DOE under NWPA, this Federal interim storage is available only as a last resort under NWPA criteria and NRC implementing regulations (10 CFR Part 53). Thus, utilities continue to develop plans for providing additional storage capacity as they approach current storage limits of their reactor pools.

When possible, utilities continue to re-rack spent fuel pools, a measure that has extended storage capacity for most reactors into the 1990's. Beyond re-racking, rod consolidation is being considered by some utilities as a means of increasing pool storage capacity. On-site dry storage of aged spent fuel in modular units is also being closely studied for meeting storage needs.

Six topical safety reports for dry storage designs are presently being considered by the NRC staff. If found acceptable by the staff, a topical report can be referenced in the license application by a utility to expedite the review of a proposed dry storage system. Final evaluation of the topical report of the Castor Ic cask design submitted by Gesellschaft fur Nuklear Service (GNS) of West Germany was completed. On May 14, 1985, a letter of approval for the topical report was issued accompanied by a safety evaluation report. This metal cask has a capacity of 16 BWR fuel assemblies. General Nuclear Systems, Inc., a partnership of GNS and Chem-Nuclear Corp., submitted a revised report on the Castor V cask design in January 1985. On September 30, 1985, a letter of approval for the topical report was issued, accompanied by a safety evaluation report. This cask has a capacity of 21 PWR assemblies and is proposed for use by the Virginia Electric and Power Company (VEPCO) at its Surry Nuclear Power Station under a license application being reviewed by the NRC staff. In the Surry case, a Finding of No Significant Impact was published in the *Federal Register* on April 18, 1985, (50 FR 15517) and an Environmental Assessment issued. Two topical reports on designs for stainless steel and lead casks with liquid neutron shields for capacities of 24 PWR and 52 BWR fuel assemblies, previously submitted by Ridihalgh Eggers and Associates, have been assumed by Mitsubishi Heavy Industries, Limited, which is actively pursuing approval of the PWR cask design topical report.

Topical reports for dry cask designs have been submitted by Nuclear Assurance Company, Westinghouse and, in September 1985, by Transnuclear. The Nuclear Assurance Company cask is of stainless steel and lead with a liquid neutron shield and has a capacity of 31 PWR fuel assemblies. The Westinghouse and Transnuclear casks are of forged steel with solid resin neutron shields and each have capacities of 24 PWR fuel assemblies. Reviews of the Nuclear Assurance Company and Westinghouse casks are underway and, review of the Transnuclear cask began in October 1985.

A topical report for dry concrete module storage of seven PWR fuel assemblies sealed in a stainless steel canister has been received from NUTECH, Inc. This report is being reviewed. It is associated with a license application for dry storage by Carolina Power & Light Company at its H. R. Robinson power plant site in South Carolina. License review of the Carolina Power & Light Company application is proceeding in parallel with the NUTECH topical report review.

Monitored Retrievable Storage

The Department of Energy (DOE) had planned to submit a proposal for monitored retrievable storage (MRS) to the Congress by June 1, 1985, as directed by the Nuclear Waste Policy Act. The MRS was to serve as a backup to the repositories, receiving and storing spent fuel and high-level waste until a repository was ready.

As a result of further analysis, DOE developed an Improved Performance Plan. In this concept, monitored retrievable storage is not considered to be a backup to a repository; rather, it is an integral component of the high-level waste disposal systems. DOE informed the Congress of its intent to submit an MRS proposal under the Improved Performance Plan in February 1986.

The Commission has consulted informally with DOE, as specified by the Nuclear Waste Policy Act, on this revised concept. The principal design idea, i.e., a large hot cell (dry) complex using sealed concrete casks for storage, has not changed. The Commission will comment on the MRS proposal, as revised, and these comments will be submitted to the Congress with the MRS proposal.

Staff is currently revising 10 CFR Part 72 to help provide the regulatory framework for licensing an MRS. (See Chapter 11.)

The NRC continued through 1985 to monitor Department of Energy quality assurance and environmental surveillance programs, including that for the West Valley (N.Y.) Demonstration Project. This photo shows West Valley engineers conducting non-radioactive tests on a slurry-fed ceramic melter, the world's largest, which will convert liquid high-level waste to durable borosilicate glass.

MATERIALS LICENSING

The NRC currently administers approximately 8,900 licenses for the possession and use of nuclear materials in applications other than the generation of electricity, or operation of a research reactor. Of these, about 300 are academic, 2,800 are medical, and 5,800 are industrial licenses. The NRC's licensing program is designed to ensure that activities involving such uses of radionuclides do not endanger the public health and safety. The agency took nearly 6,000 licensing actions during fiscal year 1985. Of these, 700 were on applications for new licenses, 3,600 concerned amendments, 1,500 were license renewals, and 200 were sealed source reviews. In addition to these NRC licenses, the 27 Agreement States administer approximately 14,000 licenses. These Agreement States have authority over such materials under regulatory agreements with the NRC (see Chapter 9).

By April 1984, most of the materials licensing program had been decentralized (see 1984 NRC Annual Report, p. 73). In April 1985, most Federal agency materials licenses were transferred to the five Regions. With the completion of this phase of a multi-year process, approximately 94 percent of the materials licensing program is now administered by the Regions.

Oversight Program

Headquarters and regional staffs continued to refine the National Program Review regimen, which was developed to assure the technical adequacy, timeliness, and consistency of the decentralized licensing program. In fiscal year 1985, a concerted effort was made to streamline the resources required to assess and upgrade the regional programs and to improve Headquarters' ability to provide technical assistance. As a result of experience gained in previous years, and because of an increased reliance on preparatory analysis and more focus on specialized reviews, fewer resources were expended than in past years.

Despite the decreased resource commitment, NMSS staff contributed to several new regulatory guides and standard review plans published by the Office of Nuclear Regulatory Research. While the regulatory guides assisted potential licensees, the standard review plans supplemented the day-to-day interchange between headquarters and regional staff. The oversight process also included monthly conference calls, and annual management seminars, reviewer workshops and visits to each Region.

An organizational change at Headquarters reflected the shift in responsibilities as a result of decentralization. In April 1985, the Material Certification and Procedures Branch was merged with the Material Licensing Branch. The newly enlarged Material Licensing Branch assumed all of the policy and licensing functions of both branches and the merger conserved resources and simplified the organizational structure.

MATERIALS LICENSEE ADMINISTERED BY NRC*
(AS OF SEPTEMBER 1985)

Licenses Administered By:

Headquarters	500
Region I	2700
Region II	900
Region III	3600
Region IV	900
Region V	300
Total NRC Licenses	8900**

*In addition to the NRC licenses, some 14,000 licenses are administered by 27 states which have authority over certain materials under regulatory agreements with the NRC.

**Totals are approximate due to almost daily fluctuation in numbers.

Consolidation of Military Licenses

For several years the NRC staff has been reviewing the possibility of consolidating licenses covering military activities involving radioactive materials. The United States Air Force (USAF) and the United States Navy expressed an interest in obtaining a consolidated license for their activities. After receiving an application from the USAF and conducting a review of their proposed consolidated program, the NRC issued a consolidated license to the USAF radioisotope program in June 1985. The new consolidated license replaced 75 individual USAF licenses, and it is anticipated that substantial administrative resources and paperwork will be saved by consolidation. NRC also anticipates an application for a consolidated license from the U.S. Navy.

Industrial Licensing

NRC-licensed radioactive materials are used by industry in such areas as industrial radiography, manufacture of gauging devices, gas chromatography, and well-logging, and also by members of the general public in various consumer products. (A more detailed description of the activities covered by NRC industrial licensing may be found in the *1981 NRC Annual Report* pp. 63 and 64.)

General Licenses. There are two types of NRC licenses for byproduct, source, and special nuclear materials—specific and general. Specific licenses are issued only to specifically named persons following application and NRC review. General licenses are effective without the issuance of license documents to particular persons. However, the manufacturer of products to be distributed to general licensees must apply to NRC for a specific license. Before issuing this type of specific license for distribution, the NRC conducts a thorough safety analysis of the product. If it meets the criteria for a generally licensed product and the regulations contained in 10 CFR 32, 40, and 70, the applicant is granted a specific license for distribution of the product to general licensees.

In 1984, NRC initiated a study to evaluate the adequacy of the existing policy pertaining to the distribution of gauges containing byproduct, source, and special nuclear materials to the public under a general license issued by the NRC. This ongoing study combines the efforts of NRC Headquarters, NRC Regional Offices, and the Agreement States. Tentative study findings indicated extensive lack of compliance with 10 CFR 31.5 requirements by generally licensed gauge-users. As a result, information notices were sent to the manufacturers, distributors, and the general licensees. The notices summarized the study findings and stressed the importance of complying with NRC regulatory requirements.

The study findings also indicated that some aspects of the general licensing policy should be reviewed. NRC is now proceeding with a rulemaking to review the existing regulations pertaining to distribution and use of generally licensed gauging devices.

Data collection related to products other than gauges which are possessed and used under a general license will be completed in 1986. This will be followed by detailed analysis to determine whether the general license policy and regulations for these products also should be changed.

Source/Device Registration. NRC and the Agreement States maintain a sealed source/device registration program which helps to expedite the licensing review process when new requests for sources or devices are received. During the report period, 155 registration sheets were issued for radioactive sources and containment devices. These registration documents require a detailed safety review of the sources and devices, and the preparation of a safety analysis for use by NRC and Agreement State reviewers in the licensing process. A computerized registry system for approved sealed sources and devices is updated twice a year, using 550 reports to NRC Regional offices and Agreement States. During the report period, approximately 100 special reports were produced for NRC and other governmental users.

Medical and Academic Licensing

An estimated 10-15 million procedures are performed each year using radioactive materials for the diagnosis or treatment of patients. Many of these procedures involve NRC-licensed materials and may be conducted in hospitals or physicians' offices. NRC-licensed materials are also used in universities, colleges, and other academic institutions in certain laboratory courses and in research programs. (See the *1982 NRC Annual Report* pp. 67-68, for more detailed description of these activities.)

Medical Users' Qualifications. On May 3, 1985, the NRC staff held a public meeting of the Advisory Committee on the Medical Uses of Isotopes (ACMUI, see Appendix 2) to consider NRC's training-and-experience criteria for physicians who use radiopharmaceuticals for diagnostic imaging procedures. Current criteria call for a six-month training program. NRC has received two proposals suggesting reduction to a four-month training program. More than 20 persons, representing various professional organizations or themselves, gave statements to the ACMUI. In general, they supported a four-month training program for persons to be engaged only in cardiovascular imaging. However, this position did not have unanimous support and the ACMUI heard a wide range of opinions. Additional time was allotted for clarifying and weighing the many complex issues involved.

The NRC staff is reviewing the transcript and public comments and developing a specific proposal that will be reviewed by the ACMUI before it is published for public comment.

Part 35 Revision. The NRC staff also led the Task Force that prepared a proposed revision of 10 CFR Part 35, "Medical Use of Byproduct Material," published for public comment. Because of the rapid evolution in the medical use of radioisotopes over the last thirty years, current requirements are spread through a variety of regulatory instruments—regulations, regulatory guides, standard license conditions and other sources. The primary purpose of the proposed revision is to consolidate these requirements. Another objective is to permit licensees to make prompt use of new safety methods and to adjust their radioisotope programs to meet changes in demand for patient care services or patient load; under the revision, they would be able to modify their procedures without NRC review and approval. However, modifications to procedures would require approval by the licensee's Radiation Safety Officer, and at a hospital, by its Radiation Safety Committee.

EVENT RESPONSE

Plan for NRC Response To Materials Contamination Incidents

In 1985, the NRC took several steps to define and improve the staff capability to respond to events which might occur as the result of use or transportation of nuclear materials. These actions included a complete revision of the NMSS Radiological Emergency Response Manual to reflect changes in procedures required by the move of the Emergency Response Center to a new building, staff training at the new center, and a transportation emergency exercise to test and demonstrate staff proficiency. The training exercise involved the NRC Region III Office and State of Illinois personnel. In addition, the staff (1) responded to public comments received on the "General Statement of Policy on NRC Response to Transportation Accidents," (2) participated in revision to "Guidance for Developing State and Local Radiological Emergency Response Plans and Preparedness for Transportation Accidents," and (3) developed and issued "NRC Response Plan for Incidents Involving Nuclear Material in Unauthorized Places."

TRANSPORTATION OF RADIOACTIVE MATERIALS

The Federal Government regulates the transportation of radioactive materials primarily through the NRC and the Department of Transportation (DOT). These two agencies have divided their regulatory responsibilities, and documented them in a Memorandum of Understanding. Shipments that occur within the United States also come under regulation by the States in certain circumstances. For international shipments, DOT is the designated U.S. authority and is responsible for implementing International Atomic Energy Agency (IAEA) standards. NRC advises DOT on technical matters.

NRC staff worked on several tasks during fiscal year 1985 designed to address transportation safety issues or to provide stability to regulatory requirements regarding the transportation of radioactive materials.

Spent Fuel Shipments

In connection with a reactor fuel reprocessing plant operated at West Valley, N.Y., in the late 1960's and early 1970's, a large quantity of spent nuclear fuel was stored at that site for future reprocessing. Subsequently, the reprocessing plant was permanently shut down and a decision was made to return the stored fuel to the nuclear power plants from which it had originally come. Return of the spent nuclear fuel to the nuclear power plants was started in 1983 and completed in 1985.

In planning for operation of their fuel reprocessing plant at Morris, Ill., the General Electric Company had supplied certain nuclear power plants with a contractual provision for taking back spent fuel from the reactor to be stored for reprocessing. Later, the plans for reprocessing spent fuel in the plant at Morris, Ill., were cancelled, but the obligation to take back and store spent fuel remained in force. In 1985, shipments of spent fuel from the Cooper Station Nuclear Power Plant in Nebraska and the Monticello Nuclear Generating Station in Minnesota were sent by rail to Morris, Ill. Shipments are scheduled to continue for several years, as casks are available for rail transport.

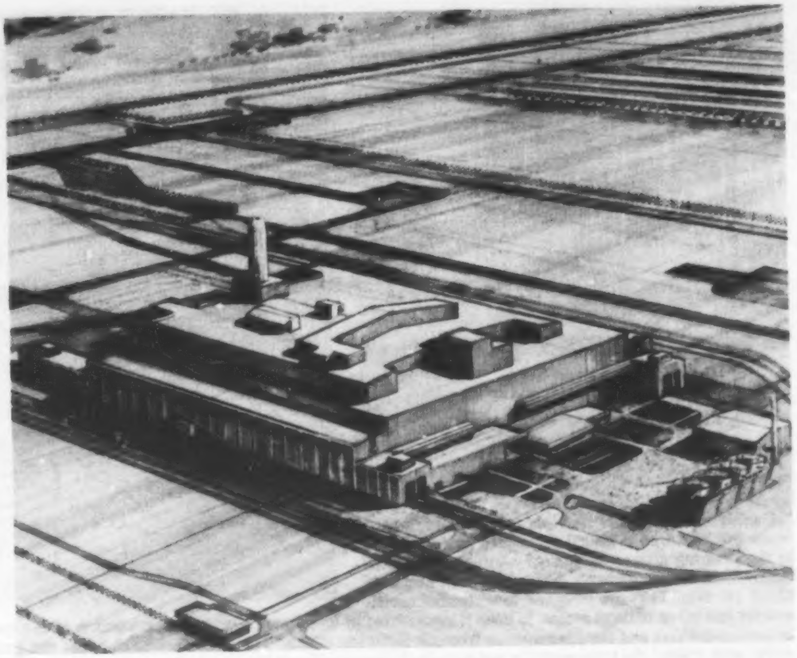
The NRC has contracted for a study of the public reactions, State and local government actions, and other repercussions from these spent fuel shipments. The study is to determine what problems were encountered, how they affected the institutions involved and members of the public, and what changes should be made in the future to forestall or alleviate the problems.

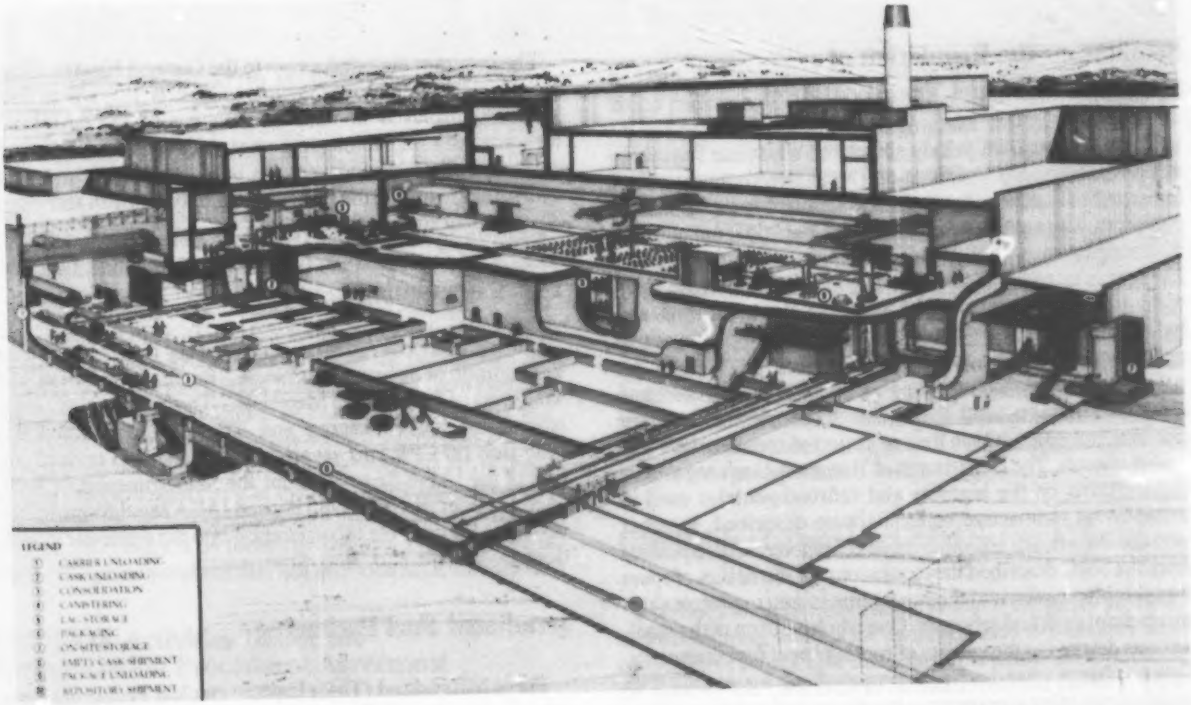
This detailed study encompasses legislative actions by states, legal actions in court, views and judgments of the State and local government agencies, and public opinion and reactions as reported in the media and expressed in interviews during the course of the study.

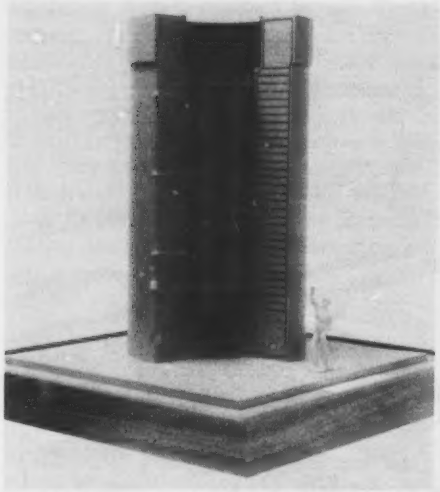
Detailed analysis of these first experiences with shipments of spent fuel from nuclear power plants is expected to help in avoiding problems as the number of such shipments increases in the future. The Monitored Retrievable Storage (MRS) Facility proposed by DOE for construction and operation by 1998 is the focal point for the first major increase in spent fuel transportation. It is expected that spent fuel shipments from nuclear power plants will be held in storage at the MRS facility until the permanent high-level waste (HLW) repository is available, and then consolidated for shipment by train to the HLW repository. Among the factors to be explicitly considered in assessing the impact of spent fuel transportation to the MRS facility and to the HLW repository are the "corridor" effects which may result should shipments from many nuclear power plants around the country be routed through a single main highway or railroad as they approach the facility. The larger quantity of spent fuel in a single cask and the number of casks to be hauled by a single train from the MRS facility to the HLW repository will also be analyzed in detail to estimate risks to public health and safety.

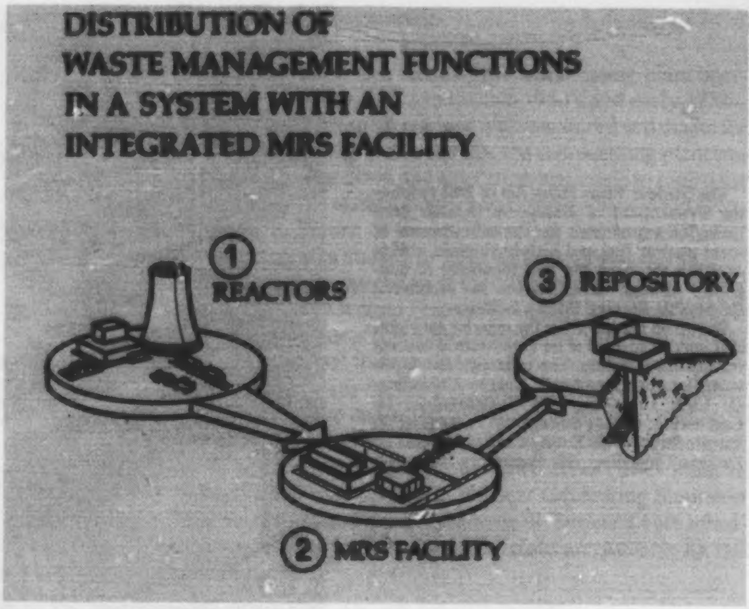
The Nuclear Waste Policy Act of 1982 requires the Department of Energy to develop deep geological repositories for the safe disposal of spent nuclear fuel and high-level waste, and to establish the need for and feasibility of temporary storage sites, known as Monitored Retrievable Storage (MRS) facilities.

Shown on this page are the exterior (at right) and interior (below) of an architectural concept for one such facility. The cutaway view below identifies work area for (1) Carrier Unloading; (2) Cask Unloading; (3) Consolidation; (4) Canistering; (5) Lag Storage; (6) Packaging; (7) On-site Storage; (8) Empty Cask Shipment; (9) Package Unloading; (10) Repository Shipment.




LEGEND
① CARRIER UNLOADING
② CASK UNLOADING
③ CONSOLIDATION
④ CANISTERING
⑤ LAG STORAGE
⑥ PACKAGING
⑦ ON-SITE STORAGE
⑧ EMPTY CASK SHIPMENT
⑨ PACKAGE UNLOADING
⑩ REPOSITORY SHIPMENT





Canistered spent fuel and encapsulated high-level waste will be stored in sealed casks such as that shown in the drawing above—steel-lined reinforced concrete cylinders 12 feet in diameter, 22 feet high and weighing about 210 tons. They are designed to withstand earthquakes and other natural events, to keep temperatures at levels which prevent degradation of the waste, and to provide shielding to protect workers and the environment from all radiation effects. The MRS facility also has the capability to receive, temporarily store, and unload the transportable metals casks being considered by some utilities for temporary spent fuel storage. The diagram at right shows the role of the MRC facility at the center of the waste disposal process, with the reactor plant at left and permanent repository at right.

Seminar on the Regulation of Spent Nuclear Fuel Transportation

The NRC and DOT jointly sponsored a seminar for designated representatives of State and local governments and of Indian tribes concerned with the regulation of spent fuel transportation. The seminar was conducted in Chicago from July 30 through August 1, 1985, and was open to the public. About 275 people attended the program, which explained the roles of the government agencies, the States, local governments, and Indian tribes in regulating the transportation of radioactive materials within their jurisdictions.

The seminar opened with the Federal agencies, the State and local governments, and the Indian tribes describing their interests and objectives in the regulation of spent nuclear fuel transportation. The Department of Transportation's regulatory requirements on the highway and railroad vehicles used in transporting radioactive materials were described, together with the inspections and other means of enforcement. Speakers from the NRC described the regulations for the design of spent fuel shipping casks, and the precautions taken to protect shipments from theft and sabotage. One whole session of the seminar was devoted to the subject of routing spent fuel shipments. Other sessions were devoted to preparedness for dealing with accidents or other emergencies involving radioactive material shipments and with inspections activities of DOT, NRC and the States to assure compliance with their regulations.

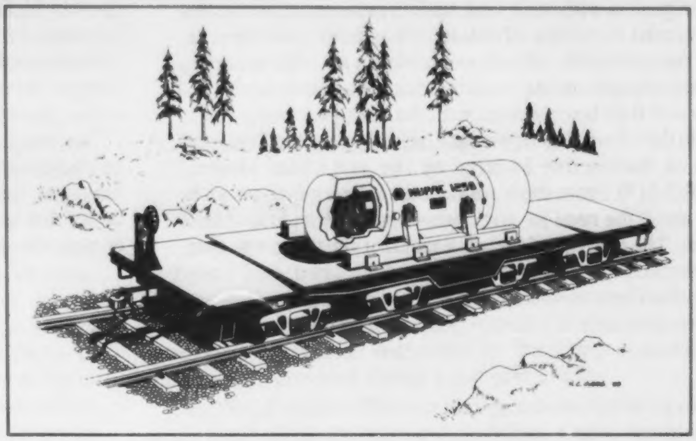
The program included a visit to the General Electric Company's Spent Fuel Storage Facility at Morris, Ill., where there were on display a number of shipping casks, emergency response vehicles, radiological emergency vans, security escort cars, and other special vehicles used by Federal and State authorities to assure the protection of public health and safety during spent fuel transportation.

IAEA Regulations

The International Atomic Energy Agency (IAEA) issued the 1985 edition of Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials." The current NRC regulations for the Packaging and Transport of Radioactive Materials (10 CFR Part 71) are based on the 1973 revised edition of the IAEA Regulation for the Safe Transport of Radioactive Materials. The 1985 revised IAEA regulations will be reviewed for possible incorporation into the domestic regulations of 10 CFR Part 71.

Irradiated Fuel Packaging

Three Mile Island (TMI) Cask. Nuclear Packaging, Inc., submitted to the NRC, on behalf of DOE, a safety analysis report for the Model No. 125-B cask. Two of these proposed rail casks are planned for the transport of the reactor core mate-



The NRC received in 1985 a safety analysis report for the Model 125-B rail cask to transport reactor core debris from the Three Mile Island Unit 2 in Pennsylvania to Idaho. Seven containers for damaged fuel can be shipped in each two-level stainless-steel, lead-shielded package.

rial from Three Mile Island (Pa.) to Idaho. The cask is a stainless steel-and-lead-shielded package. The contents are shipped dry, and two levels of containment are provided for the contents. The cask is a right circular cylinder, 65 inches in outer diameter by 207 inches in length with upper and lower impact limiters. The cavity is 511/2 inches in diameter by 192 inches in length and will contain seven fuel canisters. The gross weight of the cask and contents is 183,000 lbs.

Big Rock Point Cask. Transnuclear, Inc., submitted to the NRC, on behalf of DOE, a safety analysis report for the Model No. TN-BRP cask. DOE plans to utilize the package for the transport of 85 Big Rock Point fuel assemblies from Nuclear Service Center at West Valley, N.Y. to Idaho. The cask is a forged steel shell with an integrally-welded forged bottom, and a flanged and bolted forged top lid. The contents are shipped dry. The cask is a right circular cylinder, 831/2 inches in outer diameter by 190 inches in length with upper and lower impact limiters. The cavity is 64 inches in diameter by 171 inches in length. The gross weight of the cask and contents is 215,000 lbs.

NRC/DOE Activities Under the Transportation Procedural Agreement

The NRC/DOE Transportation Procedural Agreement published in the *Federal Register* (48 FR 51875) on November 14, 1983, remains in force. This agreement focuses on the important task of exchanging information and identifying transportation packaging issues at the earliest opportunity to assist in DOE's new cask development program. In a meeting of technical staff members on May 7, 1985, with representatives of DOT also participating, NRC staff members reported on the preliminary results of the NRC-sponsored study of the forces generated in transportation accidents and the ability of the present licensed casks to withstand these forces. DOE provided information on their plan and schedule for developing the new generation of shipping casks and the complete transportation system, including vehicles and equipment to load and unload the casks. The meeting included extensive discussion of methods of assuring that major public concerns are identified and addressed in the DOE development program. Future meetings will focus on various aspects of the development program for the new transportation casks.

Highlights of Transportation Safety Efforts

The NRC is engaged in a major study of the safety provided by its design regulations for packages used to transport large quantities of radioactive material. This study, performed for the NRC by the Lawrence Livermore National Laboratory, is to evaluate how well packages designed to meet NRC performance criteria will withstand the forces generated in severe accidents. The study is based on data from severe non-nuclear accidents that have actually occurred, supplemented by data from various package test programs. Comparing the forces resulting from severe accidents with those the casks are designed to withstand gives a measure of the degree of protection afforded by casks that conform to regulatory requirements. Accidents which produced forces in excess of those the casks are designed to withstand were studied in more detail to assess the potential for release of radioactive material from the cask. Also, the probability of such an accident's actually occurring will be evaluated and the resulting risk to the public health and safety will then be compared with the risks previously calculated in the "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes," NUREG-0170. From these evaluations, a determination can be made about the need for any changes in present performance criteria. The analysis of forces in accident situations was completed in 1985 and the first draft of the scientific report of the study was prepared.

Another important objective of the effort is to provide a document which relates regulatory performance criteria to real-world accidents in simple, straightforward language. While the significance of the NRC performance criteria can be understood by highly skilled engineers, the relationship between performance criteria and protection in accidents is not apparent to most members of the public. This has understandably been a major impediment in explaining and confirming cask safety to the general public.

The two-volume technical report is currently scheduled to be completed by the spring of 1986. Volume 1 will contain the main text, the conclusions and the recommendations and will be written for ready understanding by laymen. Volume 2 will contain the data and the scientific analyses.

Safeguards

CHAPTER 6

Under the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, the NRC regulates safeguards for licensed nuclear materials, facilities and activities to assure protection of the public health and safety and national defense and security. In this regulatory context, "safeguards " refers to measures taken to deter, prevent, or respond to the unauthorized possession or use of significant quantities of special nuclear material through theft or diversion, and to protect against radiological sabotage of certain nuclear facilities. In general, safeguards for licensed nuclear fuel facilities and nonpower reactors emphasize protection against theft or diversion of special nuclear material (SNM), while safeguards for power reactors stress protection against radiological sabotage. (SNM and Strategic Special Nuclear Materials, or SSNM, are shorthand for technical definitions of various kinds of nuclear materials, different quantities thereof, and different degrees of enrichment. In general, SSNM is highly enriched uranium or plutonium.)

During fiscal year 1985, NRC safeguards requirements were applied to 96 power reactors, 59 nonpower reactors, and 28 fuel cycle facilities. They were also applied to 171 shipments of spent fuel, 16 shipments of SNM involving more than one but less than five kilograms of highly enriched uranium, and two shipments of SNM involving five or more kilograms of highly enriched uranium.

NRC/IAEA Interaction. During 1985, the International Atomic Energy Agency (IAEA) carried out routine inspections of the Combustion Engineering Corporation's low-enriched uranium (LEU) fuel fabrication plant in Connecticut, the Arkansas-2 reactor in Arkansas, and the San Onofre-2 reactor in California. Also, the NRC continued to submit accounting data to the IAEA on a monthly basis for these facilities as well as for the LEU plants of Babcock & Wilcox at Lynchburg, Va., of Exxon at Richland, Wash., and of Westinghouse at Columbia, S.C.

In addition, the IAEA selected the General Electric LEU fuel fabrication facility at Wilmington, N.C., for reporting of material accounting information under the protocol to the US/IAEA Safeguards Agreement. The IAEA also notified the United States of its intent to select for full safeguards inspections the Westinghouse LEU fuel fabrication facility, the Turkey Point Unit 4 reactor in Florida and the Salem Unit 2 reactor in New Jersey.

In September 1985, the NRC prepared an update of the eligible facility list for application of IAEA safeguards at licensed facilities to be submitted to the Executive Branch for review and transmittal to the IAEA.

STATUS OF SAFEGUARDS IN 1985

Reactor Safeguards

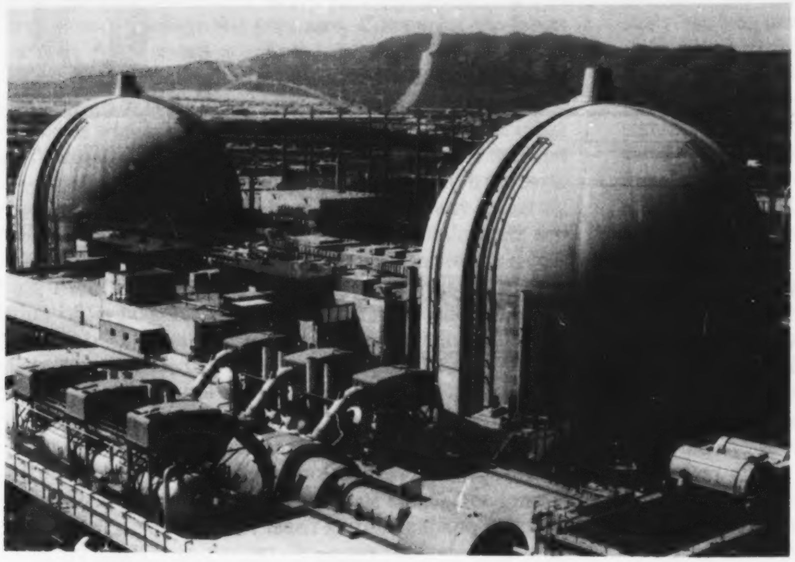
Power Reactors. The NRC continued the accelerated review of physical security plans received from applicants for licenses to operate power reactors. A total of 10 expanded comprehensive safeguards statements for the Safety Evaluation Reports was provided during fiscal year 1985.

The NRC Executive Director for Operations initiated a staff study to reevaluate the bases and guidelines used to determine the equipment and areas to be protected as vital. The study is aimed at ensuring coordination and consistency from both the safety and safeguards perspectives. The study report was scheduled to be issued in the early part of calendar year 1986.

The Regulatory Effectiveness Review (RER) program continued to evaluate the effectiveness of safeguards systems and regulations at licensed nuclear facilities and to validate the protective measures for vital equipment at power reactors. These reviews are conducted independent of NRC's regular inspection and enforcement activities and are intended to assure that safeguards programs, as implemented by licensees, are effective against the design basis threats defined in 10 CFR 73.1. During fiscal year 1985, reviews were conducted at 17 power reactors. RER's have identified both strengths and weaknesses in licensees' programs. Since the initiation of the program in 1982, nearly 30 percent of the RER findings reflect notable strengths of the security systems. Most commonly noted strengths are the quality of the security forces and their good rapport and coordination with local law enforcement agencies. The most common problem areas identified in RER reports concern vital area barriers, intrusion detection and alarm assessment systems, and protected area barriers.

The NRC continues work to resolve generic regulatory concerns and to assure timely correction of any licensee program inadequacies identified by the RER teams.

Nonpower Reactors. Although currently available information contains no indication of a specific threat aimed at a domestic licensed nuclear facility, recent acts by terrorists have shown their ability to coordinate simultaneous actions against geographically separated targets. Thus, as a matter of prudence, the NRC is considering increased security measures at nonpower reactors using high enriched uranium, as described later in this chapter. Also, in September 1985, orders were issued requiring the removal of all excess unirradiated high enriched uranium fuel from those sites which still have quantities stored on site.



The International Atomic Energy Agency (IAEA) conducted safeguards inspections at three U.S. fuel and reactor facilities during 1985 and announced its intention to select three others for inspections at a later date. Four of the six facilities are nuclear reactors, including Unit 2 of the San Onofre nuclear plant at San Clemente, Cal., shown here. The other two are fuel fabrication plants in Connecticut and South Carolina. Unit 2 of the San Onofrio station, located about midway between Los Angeles and San Diego, Cal., went into commercial operation in 1983.

Inspection and Enforcement at Reactors. During fiscal year 1985, a training program was developed to enhance the effectiveness of resident inspectors' contributions to the safeguards inspection program at power reactors. The training includes such subjects as performance of security personnel, access controls, compensatory measures, responses to alarms, and testing of detection aids. In addition, procedures were developed for inspecting physical protection required for unirradiated nuclear fuel at reactors. The inspection program for nonpower reactors was revised to: (1) reflect emphasis on theft of nuclear material, (2) incorporate new regulatory amendments, and (3) provide more detailed guidance for inspectors. (See Table 1 for a summary of inspection activity at reactors.)

Fuel Cycle Facilities. The number of licensed fuel facilities subject to NRC safeguards requirements in fiscal year 1985 remained the same as in 1984. Specifically, there were 28 such facilities, 20 of which maintained both physical security and material control and accounting systems. Four of these 20 facilities had actual holdings of formula quantities of strategic special nuclear material (SSNM), requiring the implementation of extensive physical security and material accountability measures. The remaining eight facilities were required to implement a moderate level of physical security, but were not required to implement detailed material control and accounting systems. The activities at these 28 fuel facilities include full scale production, pilot plant operations, decommissioning efforts, and the storage of sealed items. The NRC received and completed action on approximately 180 licensing matters associated with these facilities in 1985.

The effectiveness of the safeguards programs at two major fuel cycle facilities with formula quantities of SSNM was evaluated by the RER program during fiscal year 1985. Actions were initiated to correct identified deficiencies.

In May 1983, the Commission placed in abeyance a hearing that had been requested by the Natural Resources Defense Council in 1980 regarding the Nuclear Fuel Services facility at Erwin, Tenn., and laid out performance criteria for the facility's inventory differences over a two-year period. This action was taken after all parties involved had submitted a joint motion to the Commission requesting a tightening of the inventory difference limits for reinventory of the facility. The licensee met the performance criteria for the subsequent two years of operation. In accordance with the Commission Order and letter, the licensee now has the same reinventory requirements as other licensees. Action is currently underway to terminate the hearing formally.

Inspection and Enforcement at Fuel Facilities. In fiscal year 1985, new material control and accounting safeguards inspection procedures were developed for licensed nuclear fuel facilities affected by the low-enriched uranium reform amendments. These new procedures will be implemented at LEU commercial fuel facilities when their new fundamental nuclear material control plans are approved. New physical security safeguards inspection procedures also were developed for fuel facilities authorized to possess Category II or III special nuclear material. These procedures take into account changes to the regulations and apply new techniques in the inspection program. (See Table 1 for a summary of inspections conducted at fuel facilities.)

Transportation

Spent Fuel Shipments. During fiscal year 1985, NRC approved 46 transport routes from the viewpoint of protection against sabotage. One hundred thirty-five spent fuel shipments

Table 1. Summary of Safeguards Inspections Visits—FY 1985

	Number of Licensee Sites Inspected	*Number of Inspection Visits*	*Number of Violations*	*Manhours of Inspection Effort*
FUEL FACILITIES				
Formula Quantity	5	53	20	2,392
Less than Formula Quantity	8	39	13	1,851
TOTAL	13	92	33	4,243
POWER REACTORS				
Operating	86	222	121	5,653
Pre-Operating	18	36	0	1,930
TOTAL	104	258	121	7,583
NON-POWER REACTORS				
TOTAL	20	26	14	770
SHIPMENTS				
Formula Quantity	2	2	1	33
Irradiated Fuel	5	90	0	360
TOTAL	7	92	1	393
OTHER	5	6	0	82
GRAND TOTAL	149	474	169	13,071

N.B. Because of the multi-disciplinary nature of most inspections and documentation, the safeguards portion of these inspections can only be estimated.

were made over these routes. To keep the public informed about spent fuel shipment routes, NRC published, in June 1985, the fifth revision of NUREG-0725, entitled, "Public Information Circular for Shipments of Irradiated Reactor Fuel," which contains all approved routes.

SSNM Shipments. Two export shipments, each involving five or more kilograms of highly enriched uranium, were made during fiscal year 1985. Four export, three import, and 16 domestic shipments, each involving less than five but more than one, kilograms of high enriched uranium, also were made during the reporting period.

Shipment Route Surveys. In fiscal year 1985, NRC safeguards teams, each composed of two regional representatives from the Region concerned, worked with more than 165 local law enforcement agencies to conduct field surveys of routes proposed for shipments of spent fuel or SSNM. Fifty-nine routes were analyzed through 36 states, involving over 3,000 miles of travel. The NRC brochure entitled, "Information Package on Spent Nuclear Fuel Shipments for Law Enforcement Agencies" was distributed to local officials and agencies during these surveys.

Transport Inspection and Enforcement. During fiscal year 1985, the NRC continued to inspect selected domestic shipments and the domestic segments of import and export shipments of SSNM. These shipments were inspected at points of origin, in transit, during intermodal transfer and temporary storage, and at destinations. No items of noncompliance were noted. (See Table 1 for a summary of transportation inspection activity.)

Contingency Planning and Threat Assessment. Safeguards contingency plans deal with threats, thefts, and sabotage relating to licensed material and facilities. In August 1985, the NRC staff completed a review and revision of its headquarters' contingency plan in accordance with "Agency Procedures for the NRC Incident Response Plan" (NUREG-0845). Training in response plan and incident response center procedures was conducted for safeguards response team members.

Although the staff did not identify a significant change in the domestic threat environment, the NRC continued to review, in consultation with other Federal agencies, the domestic and foreign threat environments and their relationship to NRC's domestic safeguards regulations. The staff also reviewed threat-related information, on a continuing basis, to monitor any change in adversary characteristics and to assess safeguards related events associated with NRC licensed facilities and activities. The "Communicated Threat Credibility Project" continued to provide support in the form of guidance to NRC, the Department of Energy, the Federal Bureau of Investigation, and other concerned agencies for investigation of communicated threats.

The staff has begun computer-assisted analysis of safeguards events data to identify trends, patterns, and anomalies. NRC's published compilation of safeguards events data, the "Safeguards Summary Event List" (NUREG-0525) was updated in May 1985 (Rev. 10). This document contains information about safeguards-related events involving licensed nuclear material and facilities.

SAFEGUARDS REGULATORY ACTIVITIES AND ISSUES

Reactor Safeguards

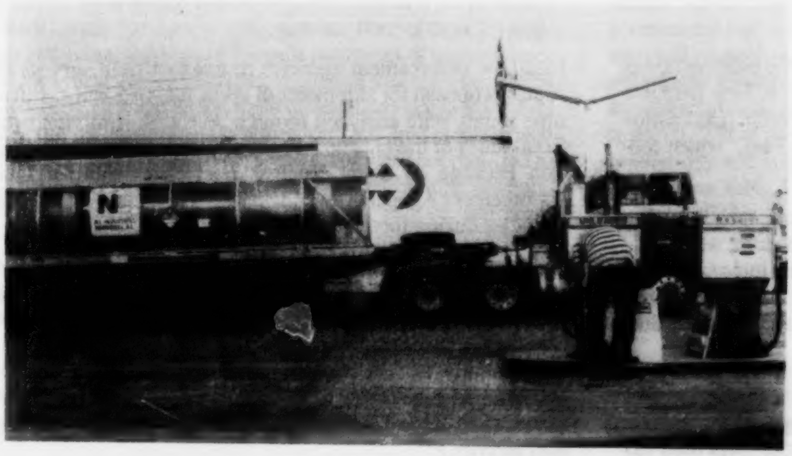
Power Reactors. Analysis of public comments on the "Insider Safeguards Rules," published in proposed form on August 1, 1984, was completed in early 1985; final requirements are under development. The purpose of this package of three related rules is to assure continuing adequacy of protection against the "insider" threat at power reactors. The cornerstone of the draft final package is the Access Authorization Rule which provides for a screening program for persons seeking unescorted access to the protected and vital areas of power reactor facilities. As proposed, the screening requirements consist of three major components: background investigation, psychological assessment, and continual behavioral observation programs. The other two rules of the draft final package clarify and refine requirements for the protection of vital equipment and requirements for physical pat-down searches of employees and visitors at protected area portals. The staff expects to submit the Insider Safeguards Rules package, revised in response to public comment, to the Commission for approval of publication in final form during late 1985.

Nonpower Reactors. The NRC is preparing a proposed regulatory amendment to require improved physical security at nonpower reactors using high enriched uranium fuel. These measures include: (1) enhanced intrusion detection capability, (2) protection against theft by an insider, (3) enhanced alarm assessment and response capabilities, and (4) removal of excess fresh fuel.

Fuel Facilities Material Control and Accounting

Low Enriched Uranium (LEU). In February 1985 the NRC published revised material control and accounting requirements for low enriched uranium in 10 CFR Part 74. The requirements are consistent with the low strategic significance of LEU and are performance oriented to provide maximum flexibility to the licensees in program design. Detailed acceptance criteria were also developed to aid in the preparation and review of licensees' material control plans.

Strategic Special Nuclear Material (SSNM). A final rule containing revised material control and accounting requirements for facilities authorized to possess and use formula quan-



Forty-six transportation routes in the United States were approved by the NRC in 1985. In June, the information circular covering shipments such as this one was revised to reflect the new routes.

tities of SSNM is in the latter stages of preparation. The final rule will take into account significant points identified in public comments and in site-specific value impact analyses performed during fiscal year 1985. The revised requirements will not apply to reactors or irradiated fuel reprocessing facilities.

Transportation

Convention on Physical Protection. The United States is a signatory of the Convention which provides for the establishment and maintenance of adequate physical security for international shipments of significant quantities of source or special nuclear material. A final rule to bring NRC regulations into accord with the Convention was issued on March 28, 1985; however, the provisions of the rule will not become effective until 21 countries ratify the Convention. As of September 1985, 13 countries had ratified.

Protection of Spent Fuel Shipments. The development of a final rule for the protection of spent fuel shipments against sabotage continued. The relationship of this rule to the safety aspects of spent fuel transportation was assessed during fiscal year 1985.

SAFEGUARDS RESEARCH STANDARDS AND TECHNICAL ASSISTANCE

During fiscal year 1985, approximately $5.1 million was spent on safeguards technical assistance and research contractual projects. Of this amount, approximately $4.4 million was spent on technical assistance projects, and the remaining $0.7 million on research projects. Some of these projects are described below.

- *Technical Assistance to Strengthen IAEA Safeguards.* This project provides technical assistance to the NRC staff in support of U.S. Government efforts to strengthen IAEA safeguards. Under the Nuclear Non-Proliferation Act of 1978, the United States is committed to continue strong support to strengthen IAEA safeguards. During fiscal year 1985, studies were undertaken to determine the relative costs, impacts, and effectiveness of adopting certain design features that have a significant influence on IAEA inspection and verification capabilities at plutonium conversion facilities.
- *Advanced Statistical Technology.* In fiscal year 1985, this project focused on demonstrating the feasibility of using sequential statistical testing techniques to satisfy the "recurring loss detection" requirements of a proposed material control and accounting rule. Candidate statistical techniques were identified, and an applications manual was prepared. A software program is under development that will be useful to the NRC and its licensees in the preparation and subsequent review of fundamental nuclear material control plans submitted in response to the revised material control and accounting requirements.

This photo was taken from the cab of a security escort vehicle accompanying a spent fuel transport en route to a laboratory. Such surveillance is continuous during all such shipments.

- *Nuclear Power Plant Vital Area Definition.* This continuing Los Alamos National Laboratory project provides systematic analyses of nuclear power plants to identify those areas and combinations of areas in which sabotage actions could result in radiological release in excess of 10 CFR 100 limits. The NRC staff, in carrying out the Regulatory Effectiveness Review Program described earlier in this Chapter, uses the results of this project to identify equipment and areas in each plant which might have to be protected as "vital," depending upon future NRC policy decisions concerning vital areas.

Safeguards Research

- *Research in Support of Licensing.* Two RES studies were completed in fiscal year 1985 to improve the technical bases for safeguards licensing. These were: (1) research to provide guidance to licensees for developing response procedures for recurring loss from testing a site specific set of material loss alarm resolution procedures for a segment of an existing nuclear material processing plant; and (2) research to update Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials." Three multi-year research projects which were initiated in prior years, continued in fiscal year 1985 to: (1) assess and update reactor vital area determination assumptions in support of the Regulatory Effectiveness Review (RER) Program and possible subsequent modification of protection strategies for vital equipment employed at nuclear power plants;

Typical physical protection features for doors opening on vital areas of reactor plants include access control by means of a key-card reader and doors, such as that shown here, locked and alarmed with balance magnetic switches.

(2) develop guidance for licensees in defining, developing, implementing, and maintaining computer managed physical security systems; and (3) quantify experimentally the magnitude and chemical/physical form of any related radioactive material which may result from sabotage of vitrified high-level waste shipments.

- *Standards Development*. Final preparation for publication of two handbooks, "The Statistical Handbook" and "The Handbook of Passive Non-Destructive Assay of Nuclear Material" continued. Also, the staff of NRC's Office of Nuclear Regulatory Research completed review for revision or withdrawal of the existing MC&A regulatory guides, to ensure that all guides are relevant to the current regulatory requirements.

SAFEGUARDS DECENTRALIZATION

Licensing functions involving review of safeguards system changes that do not decrease the effectiveness of the safeguards program, as defined in 10 CFR 50.54(p) and 10 CFR 70.32(c), (d), (e), and (g), have been transferred to the NRC Regional Offices. The responsibility for the conduct of transportation route surveys has also been transferred to the Regional Offices.

Waste Management

CHAPTER 7

The NRC's regulation of nuclear waste is managed and coordinated by the Office of Nuclear Material Safety and Safeguards (NMSS). The activities of this office include the regulation of all commercial high-level and low-level radioactive waste and uranium recovery activities. Specifically, the functions of NMSS include:

- Developing the criteria and the framework for high-level waste regulation, including the technical bases for the licensing of high-level waste repositories.
- Providing program management for NRC's responsibilities under the Nuclear Waste Policy Act of 1982 (NWPA).
- Licensing and regulating low-level waste disposal facilities and providing the technical support for such regulation.
- Providing national program management for licensing and regulating uranium recovery facilities and associated mill tailings.
- Reviewing and concurring in significant Department of Energy (DOE) decisions related to inactive mill tailings sites and the licensing of monitoring and maintenance programs for stabilized tailings piles.

Highlights of 1985

In fiscal year 1985, NRC staff carried out its responsibilities to assure that the milestones of the NWPA can be met. It is the NRC's policy that, absent unresolved safety issues, NRC will support DOE schedules for meeting NWPA requirements as set forth in the DOE Mission Plan and Project Decision Schedule. During the year, NRC provided extensive comments to DOE on its Environmental Assessments for nine potential repository sites; worked closely with DOE to resolve preliminary issues related to DOE's development of Site Characterization Plans, as well as preliminary issues related to site characterization activities; testified before Congress on implementation of DOE's Final Mission Plan, and developed comments for DOE on its Project Decision Schedule. In addition, NRC published four Generic Technical Positions on repository issues, a Format and Content Guide for Site Characterization Plans, a final technical amendment to Part 60 concerning the siting of a repository in the unsaturated zone, and a proposed procedural amendment to Part 60 concerning site characterization activities and NRC/State/Tribal participation. Significant effort was also given by the staff during the report period to conducting meetings and workshops with DOE, as well as with States and Indian Tribes, in an effort to identify and resolve issues as early as possible.

In the area of low-level waste, the disposal site licensees began amending their licenses in order to reflect 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." The NRC staff pursued Part 61 implementation in the areas of disposal site licensing guidance and greater-than-Class-C waste and also published guidance on waste form, waste classification, soil analysis and alternative waste containers.

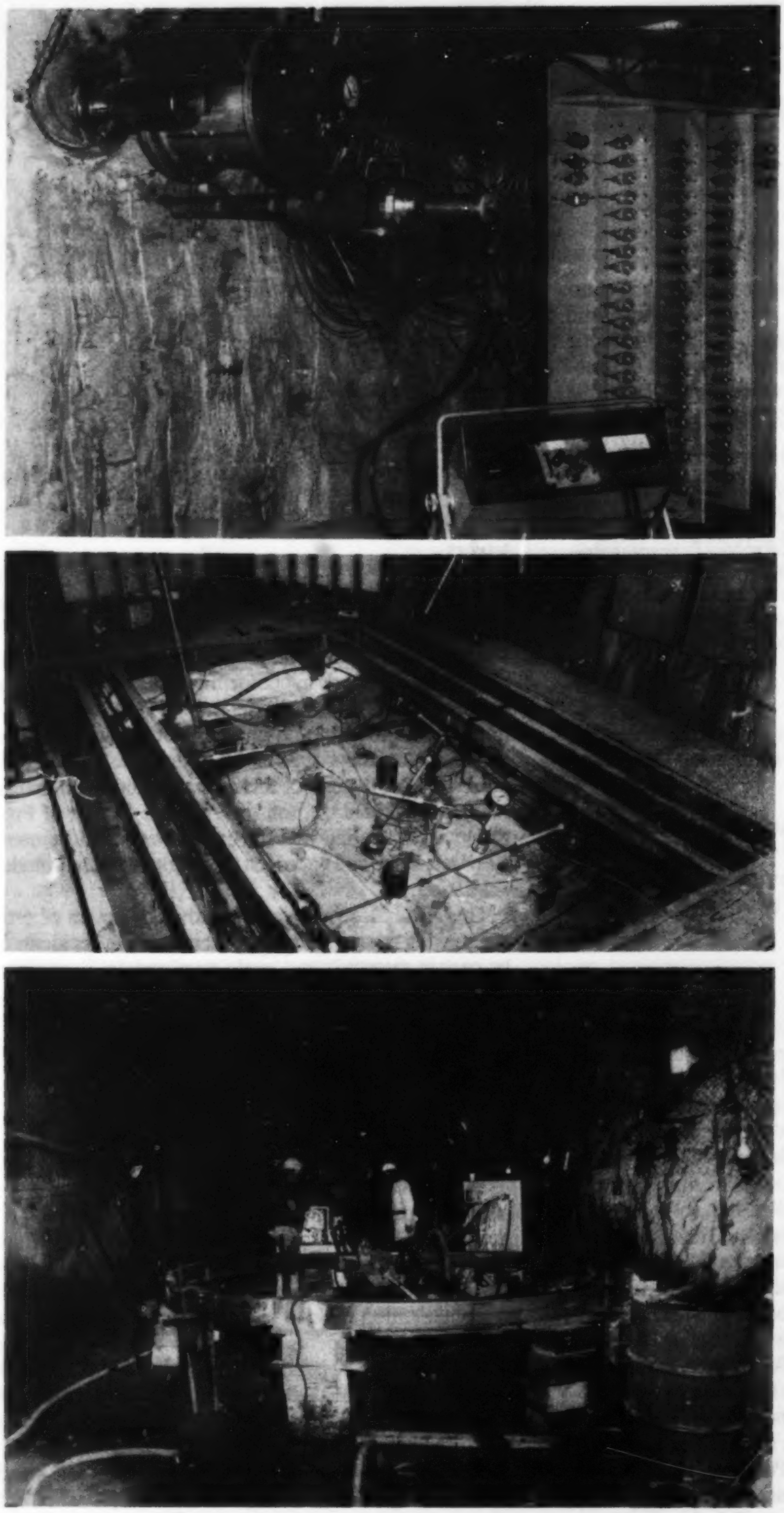
With regard to uranium recovery activities, the staff continued its involvement in the Uranium Mill Tailings Remedial Action Program at inactive sites, as required by Title I of the Uranium Mill Tailings Radiation Control Act of 1978. The staff also continued work in conforming its Part 40 regulations to the final EPA standards for mill tailings. A proposed rule conforming Part 40 to the non-groundwater requirements was published in November 1984 and the final rule was approved for publication at the end of this report period. Comments were received and analyzed on the Advance Notice of Proposed Rulemaking dealing with the groundwater requirements, and a revised scope and approach for the rulemaking is under consideration.

The NRC also published an advance notice of proposed rulemaking regarding funding for cleanup of accidents caused by certain materials and waste management licensees.

HIGH-LEVEL WASTE PROGRAM

Regulatory Development

The staff continued its rulemaking proceedings to amend the procedural and technical provisions of 10 CFR Part 60, "Disposal of High-Level Radioactive Waste in Geologic Repositories." A proposed rule was published in February 1985 to amend Part 60 to conform with NWPA requirements for site characterization reviews and NRC/State/Tribal interactions. After invited comments were received, a final rule was prepared which is to be published in fiscal year 1986. A final technical rule was published in July 1985, which amends Part 60 to include consideration of geologic repositories that may be excavated in the unsaturated zone. In addition, the staff coordinated with the EPA on its final high-level waste standards, which were published in September 1985 (40 CFR Part 191, "Environmental Standards for the Management and Disposal



The NRC staff in 1985 continued its rulemaking activities concerning the disposal of high-level wastes in geological repositories. These actions included the amending of regulation 10 CFR Part 60 to ensure its conformity with the Nuclear Waste Policy Act of 1982 (NWPA), staff coordination with the Environmental Protection Agency (EPA) on high-level waste management standards, and other moves related to the Department of Energy's (DOE) Environmental Impact Statement on the geologic repository and NRC's prospective adoption thereof. The staff will publish proposed amendments to conform Part 60 to the EPA standards in fiscal year 1986.

The photos on this page illustrate several instruments and aspects of the testing by which the DOE might evaluate the geological characteristics of candidate high-level waste repository sites. At top are shown devices used in heater tests to measure mechanical, thermal and hydrological properties of the rock in which waste might be placed. The photo at center shows the placement of instruments in this rock surface to permit the measuring of large-scale thermal and geomechanical properties of the "host" rock. And at bottom is a diamond slot cutter used to prepare the tuff medium for the conduct of all these various tests.

of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes"). The staff was reviewing Part 60 requirements against the final EPA standards at the close of the report period and will publish proposed amendments to conform Part 60 to the EPA standards in fiscal year 1986.

Two other rulemaking actions were initiated during the year. An Advanced Notice of Proposed Rulemaking to define "high-level waste" in light of the NWPA definition (which may encompass more types of highly radioactive material) was forwarded to the Commission in September 1985. Also, the staff initiated action to amend Parts 60 and 51 to conform National Environmental Protection Act-related requirements to NWPA requirements concerning NRC adoption of DOE's Environmental Impact Statement for the geologic repository. A proposed rule is expected to be published during fiscal year 1986.

Regulatory Guidance

NRC's regulatory guidance activities are directed at providing DOE with licensing information needs, acceptable methods for demonstrating compliance with Part 60 performance objectives (which are based on a "multi-barrier" approach to repository design), and acceptable methods and tests for site characterization activities. In conjunction with its regulatory guidance development, NRC staff is also developing its own tools and methodologies for evaluating DOE's assessments of repository performance.

The NRC staff continued to develop Generic Technical Positions (GTPs) and other guidance documents. The following guidance documents were published during fiscal year 1985:

- Draft GTP on In-Situ Testing
- Draft GTP on Waste Package Reliability
- Draft GTP on Design Information Needs in Site Characterization Plans
- Reg Guide 4.17, "Standard Format and Content Guide for a Site Characterization Plan"

Internal Draft GTPs were completed on Seismo-Tectonic Evaluation Methodology, Sorption, Disturbed Zone, and Groundwater Travel Time. In addition, public comments were received on the Draft GTP on Licensing Assessment Methodology.

Significant progress was made during the year in communicating to DOE the staff's view on allocating performance levels to the individual barriers of the repository design. (Part 60 sets forth performance objectives for the multiple barriers, but permits DOE to propose differing individual barrier performance levels if it provides assurance that the overall system performance objectives will be met.) An initial meeting on the subject was held between NRC and DOE in April 1985, and a second meeting was held in September 1985. The staff plans to continue working closely with DOE in this area.

Site Investigations

Section 112(b) of the NWPA requires DOE to recommend three sites to the President for characterization for the first repository, and to publish Environmental Assessments (EAs) for each of at least five nominated sites. On December 20, 1984, DOE issued draft EAs for nine sites under consideration for nomination. (The nine sites are in three different geologic media: basalt, tuff and salt.) The NRC staff conducted an extensive review of each of the EAs and provided its comments to DOE on March 20, 1985. As of the end of this reporting period, DOE plans to publish five final EAs and recommend three sites for characterization (in the three different geologic media) in December 1985.

After the three sites are selected by the President, Section 113(b) of the NWPA requires that DOE issue Site Characterization Plans (SCPs)—along with a waste form and packaging description and a conceptual repository design—to the NRC and the States and Tribes for comment. While these documents were not scheduled for issuance until spring of 1986, NRC worked closely with DOE throughout the year to help ensure that the final products will be complete and of high quality. NRC cooperation has included reviewing available data and information on the sites from investigations to date, consulting informally with DOE on its preliminary plans for site characterization, and attempting to resolve the major differences identified so far regarding the SCPs. The staff is also working with DOE to identify and resolve issues that involve long lead-time DOE commitments (e.g., exploratory shaft construction and in-situ testing) for site characterization activities before receiving the SCPs.

Quality Assurance Activities

During the year, the NRC staff made significant progress in providing guidance to DOE as to what constitutes an acceptable quality assurance program for repository purposes (including site characterization). The NWPA and 10 CFR Part 60 require that information used to support DOE's repository license application shall be subject to the Quality Assurance (QA) program set forth in 10 CFR Part 50, Appendix B, "as applicable and appropriately supplemented." Part 60 also requires DOE to describe in the Site Characterization Plans the QA program to be used to support pre-licensing activities.

In November 1984, NRC staff presented briefings at DOE Headquarters and the three DOE Repository Project Offices. The subject areas were: legal aspects of the licensing process, recent QA "lessons learned" from the reactor program, and unique features of the licensing procedures applicable to the high-level waste program. In December 1984, the staff conducted the first of a series of field visits to the DOE Project Offices. The primary reason for the visits was for NRC staff to become familiar with the details of the DOE QA program and also to clarify questions on implementation. The staff also arranged briefings at DOE Headquarters and the Project Offices on the possible application of NASA's safety, reliability, and quality assurance and management techniques in the high-level waste program. These briefings were based on NUREG/CR-4271, "Recommended Safety, Reliability, Quality Assurance, and Management Aerospace Techniques With Possible Application by the DOE to the High-Level Radioactive Waste Repository Program," published in May 1985.

Another related effort was preparation of staff technical positions on selected repository QA issues, such as qualification of old data, QA for exploratory testing, and the "Q" list. Initial drafts have been prepared and submitted to DOE.

DOE Mission Plan and Project Decision Schedule

Section 301(b)(10) of the NWPA requires that DOE submit to Congress a Mission Plan, which delineates how the activities required by the NWPA will be implemented. Section 114(e) requires DOE to prepare and update, in cooperation with affected Federal agencies, a Project Decision Schedule for those activities.

Draft Mission Plan was submitted to NRC for comment in July 1984, and NRC provided extensive comments. The Final Mission Plan was submitted to Congress on July 9, 1935. The NRC staff reviewed the final document and testified before the Senate Committee on Energy and Natural Resources on September 12, 1985, concerning its view of the Mission Plan. NRC comments in the testimony were concerned with: the Mission Plan's 27-month NRC license application review, the timing of preliminary determination, quality assurance, pre-licensing NRC-DOE consultations, and monitored retrievable storage.

The staff also provided comments to DOE on the preliminary draft and will provide comments on the Draft Project Decision Schedule. DOE issued the Draft on July 18, 1985, and NRC was to provide comments by October 1985. The Final Project Decision Schedule was expected to be issued by DOE in December 1985.

State and Tribal Interactions

The NWPA contains extensive provisions for State and Tribal participation in the repository program. It contains specific provisions for DOE's interaction with the States and Tribes and requires both NRC and DOE to provide "timely and complete" information to States and Tribes on all repository-related "determinations or plans." In addition, NRC regulations (10 CFR Part 60, Subpart C) specify a variety of mechanisms by which States and Tribes may participate in NRC's NWPA activities. It is NRC's policy to maintain close communications with the States and Tribes so that licensing issues—as well as required activities and lead times for their participation—are identified early in the process.

During the year, the staff gave numerous presentations to the States and Tribes on the role of NRC in the NWPA process. Examples include the quarterly meetings of the National Congress of American Indians, National Governors' Association meetings, Council of Energy Resource Tribes meetings, and the New Hampshire High-Level Waste Task Force established by the Governor. In addition, the staff met with the "first round" States and Tribes to discuss lead times in order to schedule interactions with them on the NRC milestones identified in DOE's Project Decision Schedule. Other NRC-sponsored meetings or presentations have included topics such as development of NRC's information management and issue management systems, exploratory shaft design and construction, seismo/tectonic investigations, and discussion of NRC's comments on DOE's Environmental Assessments.

Following NRC's preparation of its comments on DOE's Draft Environmental Assessments, the States and Tribes participated in NRC's "readiness reviews" of the comments, as well as in an NRC meeting with DOE to clarify the NRC comments. In September 1985, at the request of the Commission, the States and Tribes testified at an NRC Commission meeting on the timing of DOE's preliminary determination of site suitability (NWPA Section 114(f)).

Other Activities

During fiscal year 1985, certain other significant actions were taken to assure that potential licensing issues are identified and resolved early, so that the NRC can fulfill its statutory three-year license review obligations under the NWPA.

The NRC initiated a pilot project to demonstrate two information management systems which it hopes will facilitate license review activities and pre-licensing guidance to DOE. A Licensing Information Management System will demonstrate the feasibility of full text storage and retrieval of NRC high-level waste documents currently available in paper files in the docket control center and public document reading rooms. The volume of NRC high-level waste documentation is growing exponentially and an efficient system is necessary to meet both the needs of the technical staff and the legal need for document discovery and Freedom of Information Act request response. The staff is also demonstrating a High-Level Waste Tracking System, which is intended to identify and track the progress of licensing concerns. The system will identify by discipline (e.g., waste package, geology) the major NRC licensing concerns, including status and milestones. In addition to documenting progress toward the resolution of technical concerns, the system will assist the staff in focusing its efforts and resources on critical licensing concerns.

LOW-LEVEL WASTE PROGRAM

Regulatory Development

Since its issuance in final form in 1983, NRC staff has pursued measures to implement the rule 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." Throughout fiscal year 1985, considerable staff effort was given to developing guidance that will assist prospective licensees in the disposal site licensing process, and to updating the Part 61 waste-stream analysis methodology for "greater-than-Class-C" waste.

The NRC staff is also developing guidance for implementation of various aspects of 10 CFR Part 61 which will be useful for programs at existing and future low-level waste disposal sites. This effort includes regulatory guidance on disposal of wastes containing radioactivity greater than the Class C limits and alternative techniques for disposal of low-level radioactive waste. Finally, NRC has initiated detailed studies of low-level radioactive waste containing hazardous chemical constituents (so called "mixed wastes"). The results of these studies will be used to develop options for dealing with mixed wastes and resolving regulatory jurisdictional issues with the Environmental Protection Agency, which regulates hazardous waste.

Low-Level Waste Licensing

During fiscal year 1985, NRC staff conducted a safety and environmental review of the application to renew the NRC's Special Nuclear Material (SNM) license issued to U.S. Ecology, the licensee operating the low-level waste disposal facility in Hanford, Wash. The license renewal was expected to be issued by November 30, 1985. The renewed license will reflect adoption of substantive aspects of the new regulation, 10 CFR Part 61.

There were no new licensing activities at the Sheffield, Ill., site during the past year. The NRC has continued to evaluate the technical aspects of the operator's (U.S. Ecology) plans for site closure. The NRC staff is also working with the site owner (the State of Illinois) and DOE to examine the feasibility of site transfer, pursuant to Section 151 of the Nuclear Waste Policy Act.

Assistance to Agreement States

Throughout 1985, the NRC continued to provide technical assistance to the Agreement States (see Chapter 9). Technical assistance was given to the States of Nevada, California, Washington and Texas. The NRC has also provided assistance to licensees requesting help in developing contingency plans for waste disposal. The NRC has initiated an active outreach program as a means of providing guidance to and entering into discussions with States and Compacts regarding development of new LLW disposal sites.

At present, the Beatty site (Nev.) is expected to cease operation and close permanently, beginning in 1990. The State of Nevada has ratified language in the Rocky Mountain States LLW Compacts, and the Beatty site is committed to receive waste until 1989. The NRC staff has been working with the State of Nevada in developing an adequate closure plan for the waste disposal site in Beatty.

Work with Other Agencies

The NRC and EPA staff are working to resolve uncertainties posed to NRC-regulated activities by Federal environmental laws and regulations, such as the Resource Conservation and Recovery Act, and planned EPA standards for low-level waste disposal. The NRC staff consults with the DOE staff regarding low-level waste treatment, disposal technology development and low-level waste management for DOE-generated radioactive wastes.

URANIUM RECOVERY AND MILL TAILINGS

The NRC licenses and regulates uranium mills, "heap leaching" facilities, ore-buying stations, commercial solution mining (in-situ) operations, and uranium extraction R&D projects. The NRC also evaluates and concurs in the Department of Energy's (DOE) Remedial Action Plans for the cleanup of inactive uranium mill tailings sites and contaminated vicinity properties. The NRC Uranium Recovery Field Office (URFO), located in Denver, Colo., enhances the ability of the NRC to carry out this regulatory role by its proximity to the uranium industry and affected States.

Regulatory Development

The Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA), which was enacted to prevent or minimize environmental hazards from active and inactive mill operations, requires the Environmental Protection Agency (EPA) to develop radiation standards for mill tailings and the NRC to develop regulations for uranium recovery operations consistent with the EPA standards. The NRC promulgated its regulatory requirements for uranium mill tailings in 1980, but was embargoed from spending funds to implement its requirements until 1983, by which time EPA was mandated to promulgate its final standards. The final EPA standards were issued in October 1983. NRC is currently involved in a two-step rulemaking process to conform its regulations to the EPA standards.

In the first rule change, NRC's regulations pertaining to radiological protection and long-term stabilization of mill tailings were modified to conform to the EPA standards. The second rulemaking will incorporate provisions of the EPA standard dealing with protection of groundwater and will also address the more general provisions of the UMTRCA requiring the NRC to establish requirements generally comparable to those set by EPA for hazardous wastes under the Resource Conservation and Recovery Act (RCRA).

A final rule to complete the first rulemaking step was approved for publication by the Commission on September 30, 1985. The rulemaking work remaining for the NRC staff is incorporation of the EPA groundwater standards; analysis of public comments on an advance notice for rulemaking on these groundwater issues was completed in May 1985. A proposed rule is expected to be published in fiscal year 1986.

During fiscal year 1985, NRC staff continued work on regulatory guides, dealing with such topics as: long-term stabilization and erosion protection for mill tailings piles, bioassay at uranium mills, meteorological measurement programs at uranium facilities, and tailings pile cover material. The staff also initiated development of a Branch Technical Position to provide URFO with guidance on acceptable financial assurance mechanisms for reclamation, stabilization and long-term care of uranium milling facilities.

Licensing and Inspection Activities

Since 1984, the URFO has performed or assisted in 26 inspections of uranium recovery facilities. During fiscal year 1985, the Field Office licensed Wyoming Fuels Company for R&D solution mining; it is reviewing a new commercial in-situ license application for Everest Minerals. In other regulatory actions, the URFO staff completed three license renewals, nine major license amendments, and over 100 minor amendments to licenses.

Of the 32 uranium recovery facilities licensed at the end of fiscal year 1985, 14 were uranium mills, 2 were heap leach/ore buying stations, 12 were research and development solution mining operations, and 4 were commercial in-situ facilities.

Of the 32 licensed facilities, only three were in operation at the end of fiscal year 1985 (one uranium mill and two R&D solution mining facilities). Given the economic state of the uranium industry, very little licensing of new facilities is expected over the next few years. Therefore, much of the casework confronting the uranium recovery program will be in the areas of decommissioning and remedial activity, including remediation for groundwater contamination.

Technical Assistance to Agreement States on Uranium Recovery

Section 274 of the Atomic Energy Act of 1954, as amended, authorizes the Commission to enter into agreements with the Governor of any State providing for discontinuance of regulatory authority of the Commission with respect to source materials and byproduct materials associated with uranium recovery facilities. The NRC currently has such agreements with four states: Colorado, New Mexico, Texas, and Washington.

NRC conducts periodic reviews of the Agreement States' licensing and inspection programs to determine their compatibility with the NRC's programs in the same area. The NRC provides training and technical assistance to the Agreement States to help them fulfill their regulatory responsibilities. (See Chapter 9.) During fiscal year 1985, NRC reviewed the uranium recovery licensing programs of Colorado, Washington, New Mexico and Texas. These reviews examined the States' programs for mills, commercial solution mining facilities, and research and development solution mining facilities.

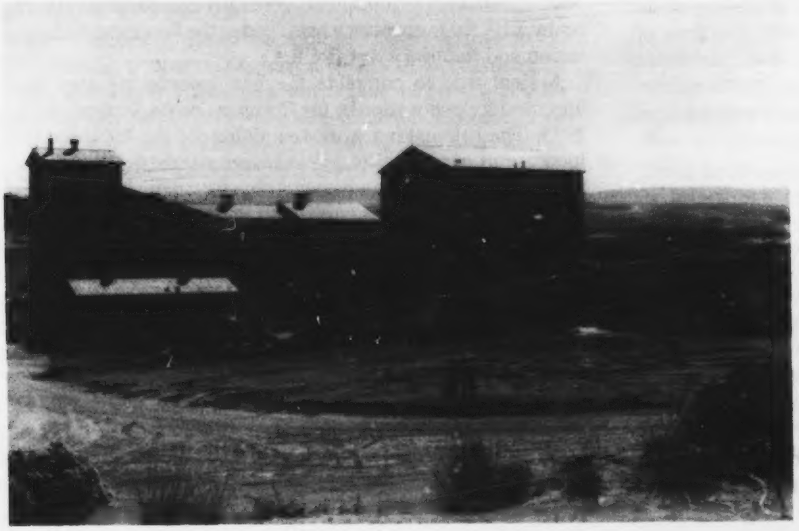
Remedial Action at Inactive Sites

The NRC has continued its involvement in the Uranium Mill Tailings Remedial Action Program (UMTRAP) at inactive mill tailings sites, as required by Title I of the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA). The NRC is a cooperating agency and is required by UMTRCA to concur in the remedial actions planned by DOE for inactive mill tailings properties.

During fiscal year 1985, the NRC staff reviewed and concurred in DOE actions connected with sites at Shiprock, N.M.; Salt Lake City, Utah; Durango, Colo.; Gunnison, Colo.; Riverton, Wyo.; Lakeview, Ore.; and Canonsburg, Pa.

The staff also reviewed and concurred in the remedial actions at several hundred contaminated vicinity properties in Edgemont, S.D.; Salt Lake City, Utah; Grand Junction, Colo.; Shiprock, N.M.; Riverton, Wyo.; Durango, Colo.; and Canonsburg, Pa.

Other accomplishments include the establishment of an NRC/DOE Memorandum of Understanding, which explains the day-to-day workings between NRC and DOE in carrying out respective agency responsibilities in the remedial action program. To promote efficiency and consistency in its actions, NRC has developed Standard Review Plans for evaluating DOE documentation and analyses of DOE proposed remedial actions.



The NRC is engaged in a two-step rulemaking action to modify its regulations for uranium mill tailings (waste from the milling process) to conform to Environmental Protection Agency standards issued late in 1983. The first phase deals with long-term stabilization of mill tailings; the second with protection of groundwater near mill tailing sites.

The photo is of a uranium mill in Utah, with the tailings piles to the right.

Inspection, Enforcement, Quality Assurance and Emergency Preparedness

CHAPTER 8

During fiscal year 1985, the Office of Inspection and Enforcement (IE) was reorganized into three program divisions, together with an enforcement staff and a program support and analysis staff. All IE activities in the areas of inspection, incident response, emergency preparedness, quality assurance and technical training continued. At the same time, greater emphasis was placed on such concerns as: the licensee/vendor interface with respect to important safety-related information and the licensees' responsibilities for the quality of their vendor-supplied equipment and services; increasing the effectiveness of inspection programs in identifying potential problems in nuclear plant operations; overall adequacy of major modifications and repairs conducted during plant outages; innovative approaches to inspections, such as examination of the operability of nuclear safety systems at operating plants; and the use of probabilistic risk assessment in focusing inspection activities on systems and components. IE continued the Performance Appraisal Team (PAT) inspections, Construction Appraisal Team (CAT) inspections, Licensee/Vendor inspections and Independent Design Inspection (IDI) efforts.

These subjects, and certain other IE Program activities, are covered in this chapter.

INSPECTION PROGRAMS

Nearly one-third of the NRC's current resources are used to develop and carry out inspection programs and procedures to verify the safety of licensees' nuclear activities and their compliance with NRC rules and regulations. The headquarters IE staff is charged with developing and promulgating comprehensive and uniform inspection procedures and policies, as well as monitoring and assessing the effectiveness and uniformity of inspection programs carried out by the five NRC Regional Offices (which are under the supervision of the NRC Executive Director for Operations). IE also conducts inspections on a national basis, as described later in this chapter. The inspection program is concentrated on those licensee activities which are most significant in terms of protection of the public health and safety. The inspection program is also structured so that increased attention is given activities outside the routine, planned licensee functions in those cases where licensee performance indicates a need for this additional NRC oversight.

Most of NRC's inspection activities are carried out by personnel located in the five Regional Offices and at reactor sites. The program basically comprises three kinds of activities. First, routine or planned inspections are conducted at all facilities in order to ensure that the safety programs established by licensees are in fact being routinely implemented and managed in a manner which will prevent a nuclear accident or unsafe condition. Secondly, NRC conducts reactive inspections in response to events or conditions at individual sites; here, the emphasis is placed upon determining the root cause of the condition, evaluating the adequacy of licensee management's response and long term corrective action to preclude recurrence, and determining whether there are generic implications for other facilities. Finally, the program affords the opportunity for each inspector to spend approximately 20 percent of available inspection time in independently pursuing and evaluating licensee programs which affect nuclear safety.

Reactor Inspection Program

The reactor inspection program is carried out by a corps of NRC resident inspectors and region-based inspection specialists. Resident inspectors are at the heart of the inspection program. They live near the sites and their offices and duty stations are on-site. While they serve in a variety of inspection functions as NRC representatives, their primary job is to observe, evaluate and report on the adequacy of licensee nuclear safety activities on a day-to-day basis. In the event of an emergency or unsafe condition, resident inspectors report to the site to assist in the collection and communication of information to NRC Region and Headquarters response teams. The region-based corps of inspection specialists supplement the basic activities carried out by resident inspectors through a variety of programmatic and technical inspections which afford an in-depth look at licensee programs.

During 1985, IE initiated several new programs aimed at increasing the effectiveness of the inspection program in identifying potential problems in nuclear plant operations. A program to review the overall adequacy of major modifications and repairs conducted during plant outages was developed and given initial testing. This new inspection procedure concentrates on a review of important design changes, their effect on original design assumptions, and on the full testing of modified equipment prior to return to service. A second inspection approach under development examines the functionality of nuclear safety systems at operating plants. In this case, one or more important systems are subjected to an in-depth safety systems functional review by a team of specialists, including both operations and design oriented personnel. Finally, inspection approaches were developed using information available from

Members of the Region III (Chicago) administrative staff and other staffers unfamiliar with the inside of nuclear power plants were taken on a tour of the Braidwood Nuclear Plant in Illinois by the NRC resident inspectors and Commonwealth Edison Company representatives.

Probabilistic Risk Assessments (PRA) in focusing inspection activities on systems and components which are important to the plant from a risk perspective. Pilot inspection programs have been initiated to test these new PRA inspection approaches with the results expected to be available in 1986.

The operating reactor inspection program is conducted by both region-based and resident inspectors. Region-based inspectors are specialists whose efforts include detailed inspections in such areas as plant operations, systems surveillance, maintenance, modifications, inservice inspection, fire protection, non-destructive testing, training, refueling, core physics testing, radiation protection, quality assurance, emergency planning, environmental protection, management systems, and security/safeguards. Resident inspectors are generalists who concentrate on day-to-day operations, event follow-up, licensee management and staff performance. This work includes close monitoring of control room activities, and of maintenance and testing carried out by the licensee, with periodic auditing of the correctness of system line-ups for nuclear systems that are important to safe operation. In addition, resident inspectors coordinate on-site activities of various NRC offices and participate in emergency exercises; they also serve as the NRC contact with local officials, the press, and the public.

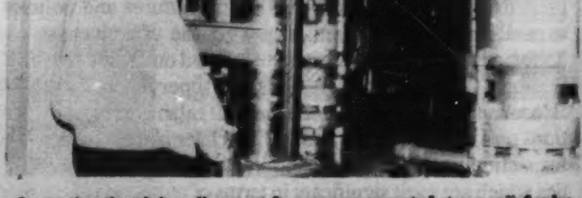
During 1985, NRC began assigning a second resident inspector at sites having a single unit operating reactor. Advantages of assigning additional resident inspectors to single-unit operating sites include increased on-site inspection coverage and increased inspection availability for coverage of non-routine events. Having at least two resident inspectors on-site also provides more flexibility in site coverage. Results of pilot programs which included the assignment of additional residents at operating sites have demonstrated the advantages of this approach.

Also during 1985, several important changes were made to the inspection program. One change involved upgrading of the inspection of licensee design change and modification programs. This change was initiated because of inspection lessons learned from a significant event involving inadequate design change and modification controls. Another change involved refocusing inspections on licensee performance in the areas of training and maintenance. The inspection procedures had previously looked mainly at the details of a licensee's programs. Other changes included the revision and upgrading of inspection programs in plant water chemistry controls, in surveillance, in control of measuring and test equipment, and in containment leak rate testing.

In 1985, the use of team inspections has increased substantially at operating sites. Team inspections have been used both (1) to focus in depth on narrow areas (e.g., plant operations), and (2) to provide a broad perspective by which to better determine root causes which cut across multi-discipline areas. Team inspections are frequently used for follow-up of a significant event and by IE headquarters offices. IE Headquarters is evaluating the innovative team inspection approaches now being employed by the Regions and is developing generic guidance on this inspection approach.

One major inspection effort continuing in 1985 is inspection of post-fire safe shutdown capability at operating and near-term operating reactor facilities. This special inspection effort, initiated in 1983, involves a team inspection of a licensee's conformance with regulations set out in Sections III.G, J and O of 10 CFR 50, Appendix R. In 1985 approximately 30 post-fire safe shutdown capability inspections were conducted. About the same level of effort is projected for fiscal year 1986.

The Emergency Preparedness Inspection Program is developed by IE and implemented by the Regional Offices. The program employs a standardized methodology to evaluate the adequacy and effectiveness of licensee emergency plan implementation and the overall state of emergency preparedness at each reactor facility. The program is accomplished



Inspecting the piping alignment for a power operated steam relief valve at the Surry nuclear plant in Virginia. Modifications were made to improve operability of the valve in handling release of steam to the atmosphere. In the photo above, the inspector is examining the gasket to be used during installation of a modified valve. These relief valves did not operate properly during the July accident and venting of steam was accomplished through the decay heat release valve.

through routine inspection and exercise observation. In 1985, the NRC monitored about 70 of the full-scale emergency preparedness exercises that are required annually. The exercises demonstrated that significant progress had been made upgrading emergency preparedness.

Also notable in this area is the NRC direct radiation monitoring network. Radiation detectors, called thermoluminescent dosimeters (TLDs), have been placed in the vicinity of all operating power reactors and those nearing completion. The TLDs are periodically replaced and analyzed to obtain an independent measure of radiation present at that location.

For reactors under construction, the region-based specialists and resident inspectors address such things as welding and nondestructive examination, civil, mechanical, electrical and instrumentation engineering, preoperational testing, emergency preparedness, and environmental protection. The resident inspector takes a more general perspective on construction activities to assure that installation of equipment and structures are accomplished in accordance with design and quality assurance requirements. The resident inspector has frequent contact with construction management personnel from the utility, architect-engineer, constructor, vendors, and contractors. He reviews procedures, observes the work, and audits quality control. He may also participate in NRC hearings, licensing meetings and public discussions. During 1985, a second resident inspector was assigned to all sites where reactors are under active construction.

Supporting the region-based and resident inspectors, NRC maintains a specially equipped mobile nondestructive examination laboratory at its Region I (Philadelphia) office.

While NRC inspection programs for reactor construction and operations cover the spectrum of activities that are important to nuclear safety, available resources permit only a limited sample of licensee activities to be examined in each of the functional areas reviewed. When deficiencies are identified through the inspection program, the NRC expects licensees to examine the deficiency in the context of all of their activities to determine whether or not it is symptomatic of a more widespread problem. Follow-up inspections by the NRC inspectors are designed to determine the adequacy of the licensee management program in this regard. Table 1 shows the number and types of licensees inspected and the number of inspections performed during fiscal year 1985.

Fuel Facilities and Materials Inspection Program

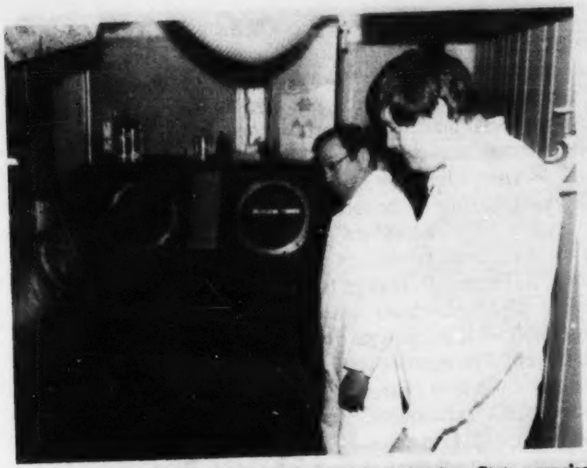
The fuel facilities inspection program covers all radiation safety and safeguards related activities at fuel production and research facilities and radiation safety at uranium mills and uranium conversion facilities. Materials licensees also receive radiation safety inspections. Included in this category are licensees whose activities involve nuclear medicine, radiography, industrial testing, well-logging, research, production and distribution of radiation sources, and the handling and storage of radioactive wastes. In addition, transportation, importing and exporting of materials, subject to NRC licensing, are included in the inspection program.

During 1985, routine inspections of materials licensees were performed in accordance with the established frequency, and special inspections were conducted in connection with allegations against licensees and reported radiation incidents. A major special inspection effort was undertaken in connection with the confiscation of about 23 curies of americium-241 from J. C. Haynes, a licensee with a small laboratory near Newark, Ohio. At the time of the recovery, the licensee was authorized to possess only about 85 mCi of the isotope in expectable contamination at the laboratory. Based on a well founded allegation, the Federal Bureau of Investigation lead a force of Federal and State agents who located about 11 curies of americium-241 in a private dwelling and about 12 curies in the laboratory. The licensee was arrested for illegal possession and use of radioactive material and for making a false statement to the NRC. Subsequently, through the auspices of the U.S. Environmental Protection Agency and its Superfund Trust, the laboratory and adjacent property were successfully decontaminated and released for unrestricted use at a cost of about $385,000.

Eleven misadministrations of radiation to patients were reported by hospital licensees in which the patient received a radiation dose substantially above that planned. As an example, a physician had intended that the nuclear medicine department perform a diagnostic scan of a patient's thyroid gland using 400 uCi of iodine-123 (half-life 13.1 hours). Due to an error, the patient was administered 10 mCi of iodine-131 (half-life 8.0 days). As a result, the patient received an unplanned radiation dose to the thyroid which NRC preliminarily has estimated to be about 8,000 rads. An NRC medical consultant is making further evaluation of the dose.

During the year, a routine inspection program was performed at all fuel facilities under NRC jurisdiction. In addition, large special inspection efforts were needed to deal with events involving the General Electric (GE) fuel fabrication plant at Wilmington, N. C., and the Nuclear Fuel Services (NFS) Navy fuel fabrication plant at Erwin, Tenn. At the GE plant, a former employee made numerous allegations against the licensee and her attorney filed a petition under 10 CFR 2.206 of the Commission's regulations requesting NRC action against the licensee. At the close of the report period, the NRC had not yet completed its investigation of the allegations.

At the NFS plant, an unexpected accumulation of enriched uranium occurred in an effluent duct in an unfavorable geometry; this raised a question about proper nuclear criticality control practice. Incidents involving inadequate control of airborne radioactive material in the plant created the possibility of unnecessary employee exposure. On May 15, 1985, the Union of Oil, Chemical and Atomic Workers (OCAW) began a work stoppage at the NFS plant. Subsequently, on August 16, the licensee began limited operation of the plant with management employees, after training of the non-striking employees and discussion of plans for limited operation with the NRC. Thereafter, OCAW petitioned the NRC, among to order, among other things, suspension of limited plant operation by management employees. The NRC was assessing the significance of the allegations in the petition at the close of the report period.



Actions taken by the Federal Bureau of Investigation, State agencies and the NRC resulted in the confiscation of about 23 curies of americium-241 from J. C. Haynes, a licensee with a small laboratory near Newark, Ohio. The licensee was authorized to possess about 85 microcuries of the isotope, for storage only. Above, NRC Regional Administrator James G. Keppler, left, and Donald Sreniawski, a Region III inspection section chief, examine equipment in the laboratory area of the facility. The glovebox at the left was used to hold the americium while the licensee conducted experiments in irradiating diamonds. Below, personnel from the Ohio Disaster Services Agency and the NRC check for contamination in an area being excavated outside the facility. The facility was decontaminated under the supervision of the NRC at a cost of about $385,000, funded by the Environmental Protection Agency's "Superfund."

Safeguards inspections of fuel facilities revealed only minor violations of NRC requirements. In 1985, the difference between the amount of special nuclear material measured during physical inventories and the "book value" of the inventory did not exceed NRC limits for these facilities.

On May 27, 1985, a new regulation (10 CFR 74) dealing with special nuclear material of low and moderate strategic significance became effective. In response to that rule, the inspection staff developed new inspection procedures for determining compliance with the new requirements.

The year 1985 saw increased spent fuel shipment activity that began in 1983. Three shipment campaigns, started in 1983, were completed in late 1984, and a fourth in July 1985. They were as follows:

- From DOE West Valley, N.Y., site to the Point Beach facility at Two Creeks, Wis.— 114 highway shipments involving 114 fuel assemblies.
- From the DOE West Valley, N.Y., site to the Dresden facility at Morris, Ill.— 31 highway shipments involving 217 fuel assemblies.
- From the General Electric facility at Morris, Ill., to the Point Beach facility at Two Creeks, Wis.— 109 highway shipments involving 109 fuel assemblies.
- From the DOE West Valley, N.Y., site to the Oyster Creek reactor at Toms River, N.J.— 33 highway shipments involving 231 fuel assemblies.

Rail shipments of spent fuel between the Monticello reactor at Monticello, Minn., and the General Electric facility at Morris, Ill., were begun in November 1984, continued through 1985 and are expected to continue over several years for a total of 30 shipments. In June 1985, shipments of spent fuel were commenced between the DOE West Valley site and the Ginna reactor at Ontario, N.Y. A total of 81 shipments is planned.

NRC inspected essentially 100 percent of the shipments at the beginning of each new campaign and reduced the frequency of inspection as experience demonstrated that there was no problem associated with a particular set of shipments. Overall, the NRC has inspected approximately 50 percent of the shipments that have been made from 1983 to the present. These inspections have found that, with a few isolated exceptions, the shipments have been in full compliance with the NRC's regulations. The few exceptions have involved a phenomenon called "weeping cask" wherein a cask that has been released for shipment based on external contamination measurements by the shipper, is subsequently found by the receiver to have external contamination above the limit. This condition is generally attributed to the weeping pores in the case of slightly contaminated water that lodged there during loading of the cask in the reactor storage pool.

The first use of the technology for dry storage of spent fuel at a commercial nuclear power plant is planned for the Surry reactor at Gravel Neck, Va. In connection with the plan, NRC inspections were conducted at the reactor site during construction of the concrete pad and at several locations in West Germany, where the storage casks are being fabricated. Additional inspections at Surry are planned during initial cask loading and storage operations after the first casks are received at the site.

Vendor Inspection Program

The Vendor Program Branch of the Office of Inspection and Enforcement conducts inspections of non-licensed organizations that provide products and services for licensed activities in order to ensure they meet applicable industry and NRC requirements. These non-licensed organizations include nuclear steam system supply and architect engineer firms, suppliers of products and/or services, testing laboratories and facilities performing equipment qualification tests, and third-party inspection organizations performing activities associated with reactor licensees. Inspections of these organizations are based on information from a variety of sources, including licensee construction deficiency and operating reactor event reports, vendor reports of product defects, allegations from members of the public pertaining to vendor activities and vendor issues identified by the NRC through its inspection programs.

In addition, during fiscal year 1985, the Vendor Program Branch conducted six inspections at nuclear reactor sites to determine whether vendor recommendations regarding the operation, modification and maintenance of vendor-supplied equipment are being appropriately reviewed and implemented by licensees. These inspections revealed significant weaknesses in the licensees' mode of receiving, considering and implementing vendor-provided information and recommendations, in cooperation with Regional Offices. Also during the report period, the Vendor Program Branch inspected nine operating reactor sites to determine, as a sample, whether safety-related equipment had been appropriately qualified for the harsh environments which could ensue from an accident. These equipment qualifications (EQ) inspection findings indicated a generally acceptable level of implementation by licensees of their EQ programs.

An NRC inspector from the Vendor Program Branch verifies a licensee's implementation of service information provided by a diesel vendor.

Since its transfer to the Office of Inspection and Enforcement from the Agency's Region IV office in fiscal year 1984, the Vendor Program Branch's objectives have expanded to include enhanced interaction with other headquarters programs rearding significant reactor safety issues, and increased efficiency of this nationwide program. These objectives are being pursued through frequent contacts with other NRC offices, through the tracking and screening of reported vendor deficiencies to identify operational safety issues and thorugh the feedback of inspection findings to Headquarters and Regional Offices. In the process, licensee responsibilities for the quality of products and services of their vendor-suppliers have been emphasized and reinforced. During fiscal year 1985, approximately 150 inspections of vendors and licensees were conducted and a number of Information Notices were issued to the industry involving deficiencies identified in vendor products or services.

APPRAISAL PROGRAMS

Systematic Assessment of Licensee Performance

The NRC program for the Systematic Assessment of Licensee Performance (SALP) seeks to evaluate the performance of each licensee with a nuclear power facility under construction or in operation in the United States on the basis of a periodic, comprehensive examination of all available data relevant to that performance.

The SALP process entails an integrated assessment based on observations as to how licensee management directs, guides and provides resources for assuring plant safety. The goal of a SALP review is to direct NRC and licensee attention toward those areas affecting nuclear safety that need improvement.

Part of the SALP assessment involves a review of the past year's Licensee Event Reports, inspection reports, enforcement history and licensing issues. Another important element is the evaluations by resident and region-based inspectors, licensing project manager and senior regional managers—all of whom are presumed to be familiar with the facility's performance. No new data are specifically generated in conducting a SALP assessment.

A SALP assessment ultimately consists of performance evaluations in a number of functional areas including plant operations, maintenance, surveillance, emergency preparedness, security and licensing issues.

The SALP program supplements the normal regulatory processes and is intended to be sufficiently diagnostic to provide meaningful guidance to utility management as to NRC concerns regarding quality and safety in plant construction and operation. The results of the program are also used by regional managers to allocate inspection resources.

Table 1. Inspections Conducted During FY 1985

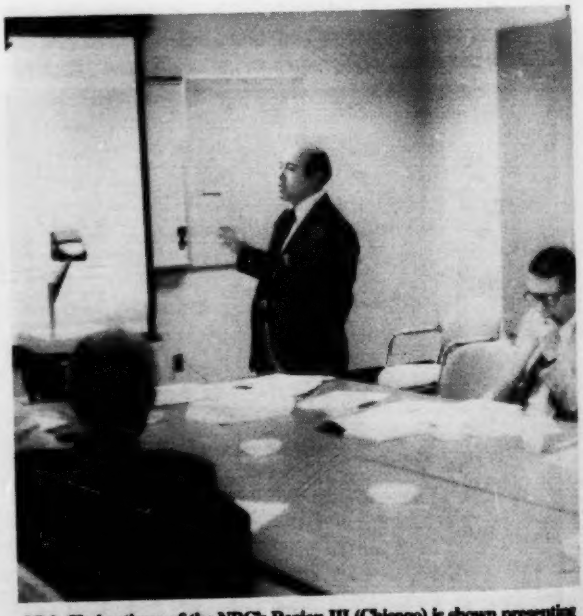
Program	*Number of Licensees Inspected*	*Number of Inspections*
Power Reactor Construction	40	1,210
Operating Power Reactors	92	3,136
Other Reactors	53	86
Fuel Facilities	42	233
Materials	2,048	2,131
Vendors (includes some licensees)	—	150
Others, including shipments	96	98

Appraisal Teams

The Performance Appraisal Team (PAT) is a group of experienced inspectors who conduct comprehensive inspections of operating reactor facilities' management control systems and related performance to determine their adequacy. The team focuses on such selected areas of plant activities as operations, maintenance, surveillance testing, design change and modification, and training. The PAT inspections of operating reactors provide an independent check on regional inspection effectiveness, assess the adequacy of headquarters program guidance, and judge the effectiveness of the nuclear industry's Institute of Nuclear Power Operations (INPO).

PAT activities over the last year included a large team inspection at the Cooper Nuclear Station (Neb.) in October 1984 and at Maine Yankee in June 1985. Also implemented were smaller scale inspections (two or three inspectors) done at San Onofre (Cal.), McGuire (N.C.), and Susquehanna (Pa.) in March 1985 and at Ft. St. Vrain (Colo.) and D.C. Cook (Mich.) in August 1985. The smaller inspections were directed at only one or two areas of facility activities and involved a more detailed technical review. As discussed previously, the PAT inspectors were involved in the development of a new team inspection approach that concentrates on the functionality of selected safety systems at operating plants. The first inspection of this type was conducted at Turkey Point (Fla.) in September 1985. Lessons learned from early trial of the new inspection approach will be factored into future PAT inspections.

In 1985, the Construction Appraisal Team (CAT) inspection program, similar in purpose to the PAT inspection program,



Nick Chrisotimos of the NRC's Region III (Chicago) is shown presenting the Region's Systematic Assessment of Licensee Performance (SALP) report to officials of the Detroit Edison Company. The SALP process is designed to alert the NRC and licensees to problem areas affecting nuclear safety. A SALP assessment consists of performance evaluations in a number of functional areas, including plant operations, maintenance, surveillance, emergency preparedness, security and licensing issues.

was continued; the goal of conducting four CAT inspections per year was exceeded by one in 1985. The primary purpose of a CAT inspection is to evaluate the design controls, construction practices, and as-built conditions at nuclear plants under construction. A CAT inspection also provides an assessment of Regional Office implementation of the IE inspection program and monitors the progress of the INPO construction project evaluation program.

CAT inspections were conducted during the report period at Millstone 3 (Conn.), Clinton (Ill.), Byron 2 (Ill.) and at South Texas. The inspections turned up deficiencies in fabrication, installation and testing. Examples include electrical wiring of motor operated valves that was not as shown on installation drawings; ASME Code-required radiographs that were not available for selected components; material traceability that was not maintained, especially for fastener materials; and unsatisfactory control of design and installation documents.

THE ENFORCEMENT PROGRAM

The purpose of the NRC's enforcement program is to protect public health and safety by ensuring that licensees comply with regulatory requirements. The program is carried out under the revised enforcement policy (10 CFR Part 2, Appendix C, March 8, 1984), which calls for strong enforcement measures to encourage compliance and which prohibits operations by any licensees who fail to achieve adequate levels of protection.

The NRC uses three types of enforcement actions, described in detail in earlier annual reports (see the 1980 NRC Annual Report, p. 144). In summary, Notices of Violations are issued for all instances of noncompliance with NRC requirements. Civil penalties are issued in case of significant or repetitive noncompliance or when a Notice of Violation has not been effective. Orders to cease and desist operations, or to suspend, modify or revoke licenses are issued to cover extremely serious cases.

Certain headquarters enforcement functions have been regionalized. The Regional Administrators have always been authorized to issue Notices of Violation not involving civil penalties. They are also authorized to issue proposed civil penalties, with the concurrence of the Director of the Office of Inspection and Enforcement. The Director of the Office of Inspection and Enforcement, however, remains responsible for all enforcement decisions and issues orders including those imposing or proposing civil penalties.

Table 2 provides a listing and brief summary of the 90 civil penalty actions taken during fiscal year 1985. Eighty-one civil penalties totaling over $3.4 million were proposed during fiscal year 1985. With some cases still pending and some of the penalties remitted or mitigated, a total of $2,332,675 in penalties was collected at the close of the report period. Some of these were civil penalties originally proposed in fiscal year 1984.

Table 3 provides a description of the 18 enforcement orders issued during fiscal year 1985.

QUALITY ASSURANCE

Quality Assurance Program Plan

The staff completed and forwarded to the Commission (SECY 85-65) a plan to implement the recommendations of the Quality Assurance (QA) Report to Congress (NUREG-1055) as modified by public comments and ACRS (see Chapter 2) and Commission guidance. Four major areas of the Quality Assurance Program Implementation Plan receiving staff emphasis are inspection programs, standards development, QA for waste management, and design inspection activities. Activities and accomplishments in these areas are described below.

NRC's Inspection Programs For Assurance of Quality

Readiness Reviews. "Readiness Reviews" were identified as a topic for further analysis in the QA Report to Congress. A Readiness Review is a formal assessment of the licensee's readiness to construct or operate a nuclear power plant. It is a comprehensive evaluation of the licensee's program for design, construction, and preoperational activities, and it examines the licensee's performance in meeting its commitments and regulatory requirements. Two such programs were initiated by utilities during fiscal year 1985. Readiness Reviews are being performed at Georgia Power Company's Vogtle Unit 1 and Washington Public Power Supply System's WNP-3. The Vogtle plant and WNP-3 are over 75 percent complete. Vogtle is under construction and WNP-3 is in a deferred status.

The Vogtle Readiness Review began in March 1985 with the submission to the NRC of a report which evaluated the safety-related concrete work completed to date. More than twenty such modules, covering all aspects of the Vogtle construction project, will be submitted for NRC review in conjunction with the Vogtle Readiness Review. In August 1985, the NRC completed its review and inspection and formally accepted the licensee's assessment of its performance in meeting commitments and regulatory requirements for safety-related concrete. All other aspects of the plant will be covered in the remaining 19 reports. This is the first time the NRC has undertaken incremental acceptance of a licensee's construction work. As of September 30, 1985, the Vogtle Readiness Review Program was approximately 20 percent complete.

The NRC also agreed to participate in a Readiness Review Program for WNP-3. The program consists of two phases. Phase One is a review of design and engineering, maintenance and preservation, and construction completed to date. The NRC had accepted the licensee's proposed program for preservation and maintenance of the work completed when the project went into a deferred status in 1983. The NRC is currently appraising a proposed program for review of the WNP-3 design and for assessing the quality of all work completed prior to work stoppage. Phase Two would begin after the restart of construction and is expected to be patterned after the Vogtle Readiness Review Program.

Table 2. Civil Penalty Actions During FY 1985

Licensee	*Amount*	*Reason*
Mississippi Power and Light Company (Grand Gulf) EA 84-23	$500,000 proposed FY 85; Pending	Violations based on deficiencies in the training program which involved three sets of RO and SRO license applications which contained material false statements, the failure to notify NRC of errors in the license applications, and the failure to correct them once the error became known to the licensee.
Commonwealth Edison Co. (Dresden) EA 84-24		Failure to use effective compensatory measures to control access into the protected area, and actions of licensee and contractor personnel in deliberately circumventing plant procedures in an attempt to expedite repair activities on the radwaste solidification system. The civil penalty was increased due to careless disregard for requirements evidenced by the failure of supervisory employees to notify the security organization of the violation. $140,000 proposed FY 84; $130,000 imposed and paid in FY 85.
Southern California Edison Company (San Onofre) EA 84-34	$250,000 proposed and $125,000 imposed in FY 84; paid in FY 85	Violations involved exceeding a technical specification limiting condition for operation requirement involving an Engineered Safety Feature System.
Duke Power Company (McGuire) EA 84-37	$40,000 proposed and imposed in FY 84; paid in FY 85	Violation involved a failure to implement adequate independent verification which resulted in a mispositioned valve.
University of Pennsylvania Philadelphia, PA EA 84-50	$4,000 proposed and imposed in FY 84; paid in FY 85	Violations involved a programmatic breakdown in management oversight and control of the radiation safety program which involved an exposure to a licensee employee in excess of regulatory limits, failure to maintain control of licensed material, failure to perform thyroid bioassays, failure to use syringe shields, failure to perform adequate evaluations of airborne effluents, and excessive radiation levels in unrestricted areas.
Virginia Electric & Power Company (North Anna & Surry) EA 84-57	$40,000 proposed and paid in FY 85	Violations involved inoperable reactor head vent system at North Anna 2 and Surry 1 and 2. Additional violations without civil penalty involved material false statements that were made regarding the status of the reactor head vent system at Unit 2 and the status of the maintenance and testing procedures for the reactor trip breakers at Units 1 and 2.
Georgia Power Company (Hatch) EA 84-59	$60,000 proposed in FY 84; imposed and paid in FY 85	Failure to adequately control access to the protected area. The civil penalty was increased because the violation represented the second failure to control access to the plant within the past year.
Nuclear Fuel Services, Inc. Erwin, TN EA 84-60	$100,000 proposed in FY 84; $50,000 imposed and paid in FY 85	Failure to maintain Material Access Area barriers in an effective and reliable condition. The civil penalty was originally increased because of multiple examples of the violation. Civil penalty was subsequently decreased due to extensive corrective actions.
Community Hospital of Anderson Anderson, IN EA 84-65	$4,000 proposed in FY 84; imposed and paid in FY 85	Violation involved the licensee's failure to implement effective management control over the radiation safety program and the falsification of records that NRC requires be maintained.

Table 2. Civil Penalty Actions During FY 1985
(continued)

Licensee	*Amount*	*Reason*
Arkansas Power and Light Company (ANO) EA 84-66	**$40,000 proposed in FY 84; $20,000 imposed and paid in FY 85**	**Failure to conduct an adequate quality assurance program relating to receipt inspections involving procurement of fasteners to ASME code requirements.**
Syncor International Corp. Sylmar, CA EA 84-73	**$8,500 proposed in FY 84; imposed and paid in FY 85**	**Distribution of radiopharmaceuticals contaminated with molybdenum-99 resulting in at least sixteen patients receiving contaminated doses of technetium-99m in excess of regulatory limits.**
Mississippi Power & Light Company (Grand Gulf) EA 84-75	**$125,000 proposed in FY 85; Pending**	**Violations is based on five alleged material false statements related to development of technical specifications governing operation of Unit 1.**
University of Connecticut Storrs, CT EA 84-80	**$2,500 proposed in FY 84; paid in FY 85**	**Violations involving failure to properly secure licensed materials. Similar violations were identified in a previous inspection and corrective actions were not sufficient to preclude its recurrence.**
Commonwealth Edison Co. (Byron) EA 84-81	**$40,000 proposed and paid in FY 85**	**Violation involved a material false statement that was made regarding source inspections of safety-related equipment for the Byron site.**
Union Carbide Corporation Grand Junction, CO EA 84-84	**$5,000 proposed in FY 84; paid in FY 85**	**Multiple violations representing a breakdown in management oversight and control of licensed activities. Although the civil penalty could have been increased due to two previous similar violations, the licensee's prompt and extensive correction action negated the potential increases.**
Research Medical Center Kansas City, MO EA 84-85	**$2,500 proposed and paid in FY 85**	**Multiple violations involving (1) failure to conduct adequate surveys, (2) failure to evaluate intake of radioactivity by an individual in a restricted area, (3) failure to perform bioassays, and (4) failure to perform personnel monitoring.**
Kansas Gas & Electric Company (Wolf Creek) EA 84-87	**$64,000 proposed in FY 84; Pending**	**Discrimination against a member of the Quality Assurance/Quality control organization.**
Minnesota Mining and Manufacturing Company St. Paul, MN EA 84-90	**$250 proposed in FY 84; paid in FY 85**	**Storage of licensed materials in an unrestricted area resulting in loss of the materials. The civil penalty was mitigated due to the licensee's prompt and extensive corrective action.**
Duke Power Company (Catawba) EA 84-93	**$64,000 proposed in FY 85; Pending**	**Violation of 10 CFR 50.7 involving a QC welding inspector.**
Toledo Edison Company (Davis-Besse) EA 84-95	**$90,000 proposed and paid in FY 85**	**Violations involving the licensee's inability to recognize design basis and technical specification requirements, to ensure appropriate 10 CFR 50.59 reviews, and to take effective corrective actions once problems were identified.**
Iowa Electric Light & Power Company (Duane Arnold) EA 84-96	**$25,000 proposed and paid in FY 85**	**Violation involved a personnel error which rendered both trains of the Standby Liquid Control System inoperable.**
Union Electric Company (Callaway) EA 84-97	**$25,000 proposed, imposed and paid in FY 85**	**Violation involved plant operation in Mode 4 with both Containment Spray Systems inoperable.**
Yale University New Haven, CT EA 84-99	**$1,250 proposed and paid in FY 85 ($178.57 subsequently returned to licensee)**	**Violations involved failure to properly secure licensed materials during transportation and failure to perform adequate health physics surveys. One violation was subsequently withdrawn.**

Table 2. Civil Penalty Actions During FY 1985
(continued)

Licensee	*Amount*	*Reason*
Globe X-Ray Service Tulsa, OK EA 84-103	$5,000 proposed and paid in FY 85	Violation involved the failure of the licensee's radiographers to adequately control the security of licensed material in unrestricted areas and the failure to take safety precautions during field radiographic operations.
Florida Power Corp. (Crystal River) EA 84-104	$50,000 proposed, imposed and paid in FY 85	Violation involved improper barriers for vital equipment.
American Electric Power Service Corp. (DC Cook) EA 84-105	$50,000 proposed, imposed and paid in FY 85	Violations involved inoperability of both trains of the Unit 1 ESF Ventilation Exhaust System, inoperability of both Motor Driven Auxiliary Feedwater Pumps, and inoperability of the Turbine Driven Auxiliary Feedwater Pump.
Tennessee Valley Authority (Browns Ferry) EA 84-106	$50,000 proposed FY 84, imposed and paid in FY 85	Violation involved several examples of failure to install vital area barriers and implement access control measures around vital equipment.
Kansas Gas & Electric Co. (Wolf Creek) EA 84-107	$75,000 proposed and paid in FY 85	Violation involved a significant breakdown in the licensee's program for the inspection and correction of defective safety-related structural steel welds.
Tennessee Valley Authority (Browns Ferry) EA 84-108	$100,000 proposed and paid in FY 85	Violations related to a core spray overpressurization event.
Pennsylvania Power & Light co. EA 84-109	$50,000 proposed and paid in FY 85	Violation involved an event in which a complete loss of all low pressure Emergency Core Cooling Systems occurred.
Trans-Eastern Inspection Services Washington, PA EA 84-110	$5,000 proposed, imposed and paid in FY 85	Violation involved the exposure of two individuals to radiation in excess of regulatory limits.
Beth Israel Hospital Boston, MA EA 84-113	$1,250 proposed and paid in FY 85	Violations represented a significant programmatic breakdown in management oversight and control of the radiation safety program.
Connecticut Yankee Atomic Power Co. (Haddam Neck) EA 84-115	$80,000 proposed and paid in FY 85	Violations involved the reactor refueling cavity seal failure which drained approximately 200,000 gallons of borated reactor coolant water to the containment floor in approximately 20 minutes.
Tennessee Valley Authority (Sequoyah) EA 84-119	$112,500 proposed and paid in FY 85	Violations involved a seal table leak and thimble tube ejection event.
Florida Power & Light Co. (Turkey Point) EA 84-121	$25,000 proposed and paid in FY 85	Violation involved a failure to maintain operability of the Intake Cooling Water System as required by technical specifications.
Omaha Public Power District (Ft. Calhoun) EA 84-122	$25,000 proposed, $21,425 imposed and paid in FY 85	Violations involved failure to maintain adequate access control at protected and vital area barriers.
Commonwealth Edison Co. (Quad Cities) EA 84-123	$50,000 proposed, imposed and paid in FY 85	Violation involved a Unit 1 licensed operator leaving his assigned position unattended with the unit operating for a period of approximately 15 minutes.
St. Luke's Episcopal Hospital Ponce, PR EA 84-125	$2,500 proposed and paid in FY 85	Violations involving inadequate control and oversight of the radiological safety program.
Nuclear Fuel Services, Inc. Rockville, MD EA 84-128	$20,000 proposed in FY 85; Pending	Violation involved the accumulation of uranium-bearing solids in process equipment above specified limits.
Duke Power Co. (McGuire) EA 84-130	$50,000 proposed, imposed and paid in FY 85	Violation involved the failure of the Upper Head Injection (UHI) accumulator system isolation valves to close at the required UHI accumulator water level.

Table 2. Civil Penalty Actions During FY 1985
(continued)

Licensee	*Amount*	*Reason*
Applied Health Physics, Inc. Bethel Park, PA EA 84-131	$1,000 proposed, imposed and paid in FY 85	Violations of numerous transportation requirements which were identified by a representative of the State of Washington during an inspection conducted at the U.S. Ecology, Inc. burial site in Richland, Washington.
Nebraska Public Power District (Cooper) EA 84-132	$25,000 proposed and paid in FY 85	Violations involved inadequacies in the performance of surveillance tests of the unit batteries.
Tennessee Valley Authority (Browns Ferry) EA 84-136	$150,000 proposed and paid in FY 85	Violations involved failures to comply with technical specifications and procedural requirements associated with a Unit 3 reactor startup.
GPU Nuclear Corporation (TMI-2) EA 84-137	$64,000 proposed in FY 85; Pending	Violation involved acts of discrimination against a contractor employee for raising safety concerns associated with the TMI-2 polar crane refurbishment in 1983.
North American Inspection, Inc. Laurys Station, PA EA 85-01	$5,000 proposed and imposed in FY 85; Pending	Violations involving inadequate management control and oversight of the radiological safety program.
Nuclear Fuel Services, Inc. Erwin, TN EA 85-03	$18,750 proposed and paid in FY 85	Violations involved the failure to perform evaluations of employee exposures and surveys of airborne activity.
Dravo Corporation Marietta, OH EA 85-05	$2,500 proposed and paid in FY 85	Violation involved an individual not authorized by the licensee's license, nor technically qualified in accordance with 10 CFR 34.31(a), using licensed material in the performance of radiography and acting as a radiographer.
Schlumberger Technical Corp. Houston, TX EA 85-06	$250 proposed and paid in FY 85	Violation involved the loss of an injection tool containing liquid I-131 for several days.
A-1 Inspection, Inc. Evanston, WY EA 85-08	$500 proposed and paid in FY 85	Violations involving an overexposure, failure to report, and failure to comply with transportation requirements.
Jackson Laboratory Bar Harbor, ME EA 85-09	$500 proposed and paid in FY 85	Violations involved inadequate oversight and control of the radiological safety program during research activities.
Louisiana Power & Light Co. (Waterford) EA 85-10	$130,000 proposed in FY 85; Pending	Violations involved quality control issues identified during inspections conducted to evaluate allegations received in 1983.
Crane Co. St. Louis, MO EA 85-11	$5,000 proposed and paid in FY 85	Violation involved the failure to control radiation levels in an unrestricted area resulting in the exposure of several members of the general public, but not in excess of NRC limits.
Allied Chemical Co. Metropolis, IL EA 85-12	$5,000 proposed and paid in FY 85	Violation involved the excessive intake of radioactive material by a worker.
Babcock & Wilcox Co. Lynchburg, VA EA 85-13	$2,500 proposed and paid in FY 85	Violation involved the failure to make surveys or evaluations, or to follow a Radiation Work Permit.
United Inspection, Inc. Tulsa, OK EA 85-14	$2,500 proposed and paid in FY 85	Violation involved a significant breakdown in the radiation safety control of licensed activities that provided a potential for unnecessary radiation exposure to licensee employees and members of the public.
Vess Beverages Maryland Heights, MO EA 85-15	$500 proposed and paid in FY 85	Violation involved the improper disposal of licensed radioactive material.
Swank-Metacon Systems Co. Pittsburgh, PA EA 85-16	$1,250 proposed and paid in FY 85	Violation involved a gauge which was located at a facility of one of the licensee's customers.

Table 2. Civil Penalty Actions During FY 1985
(continued)

Licensee	*Amount*	*Reason*
James River Corporation Easton, PA EA 85-19	$250 proposed and paid in FY 85	Violation involved the improper disposal of licensed radioactive material.
Public Service Electric & Gas Co. (Salem) EA 85-22	$50,000 proposed and paid in FY 85	Violations involved deficiencies in the training of emergency personnel and failure by management to correct deficiencies in the Emergency Preparedness Program.
Commonwealth Edison Co. (LaSalle) EA 85-26	$25,000 proposed and paid in FY 85	Violation involved the inoperability of both trains of the Standby Gas Treatment System. Civil Penalty was mitigated 50% because of unusually prompt and extensive corrective action taken by the licensee.
Kansas Gas & Electric Co. (Wolf Creek) EA 85-27	$25,000 proposed and paid in FY 85	Violations involved weaknesses in the execution of the test program. Civil penalty was mitigated 50% because the licensee took extensive corrective actions.
Duquesne Light Co. (Beaver Valley) EA 85-28	$50,000 proposed and paid in FY 85	Violation involved the licensee's failure to maintain containment integrity.
Sego Well Services Cambridge, OH EA 85-33	$500 proposed and paid in FY 85	Violations involved a breakdown in the management oversight and control of licensed activities.
Barnert Hospital Paterson, NJ EA 85-35	$2,500 proposed and paid in FY 85	Violations involved a breakdown in the management oversight and control of licensed activities.
Tennessee Valley Authority (Browns Ferry) EA 85-38	$50,000 proposed and paid in FY 85	Violation involved vital equipment which was not afforded the level of protection specified in the Physical Security Plan.
Philadelphia Electric Co. (Peach Bottom & Limerick) EA 85-42	$75,000 proposed and paid in FY 85	Violations involved inadequate management control of security and health physics activities performed by contractors at both sites.
St. Thomas Hospital St. Thomas, VI EA 85-45	$2,500 proposed and paid in FY 85	Violations involved inadequate oversight and control of the radiation safety program.
American Can Co. Greenwich, CT EA 85-47	$500 proposed in FY 85; Pending	Violation involved the unauthorized removal of radioactive material from an unrestricted area.
Tennessee Valley Authority (Browns Ferry) EA 85-51	$150,000 proposed and paid in FY 85	Violations involved the failure of the licensee to meet technical specification requirements for reactor vessel water level instrument operability and to take adequate corrective action for a similar previous problem.
Commonwealth Edison Co. (Byron) EA 85-52	$25,000 proposed in FY 85; Pending	Violation involved a failure to adequately implement compensatory measures to control access into a vital area.
Combustion Engineering, Inc. Windsor, CT EA 85-54	$5,000 proposed and paid in FY 85	Violation involved the failure to make necessary surveys and evaluations which resulted in an extremity exposure to one employee in excess of regulatory limits.
Frances Mahon Deaconess Hospital Glasgow, MT EA 85-58	$2,500 proposed in FY 85; Pending	Violation involved the use of licensed material by an unauthorized user.
Advanced Medical Systems, Inc. Geneva, OH EA 85-60	$6,250 proposed in FY 85; Pending	Violation involved significant weaknesses in management control of the radiation protection program.
Geo-Mechanics, Inc. Elizabeth, PA EA 85-62	$500 proposed and paid in FY 85	Violation involved a breakdown in management controls over licensed material.
Tennessee Valley Authority (Watts Bar) EA 85-64	$100,000 proposed and paid in FY 85	Violation involved the status of control room design modifications as required by Appendix C to the Watts Bar Safety Evaluation Report.

Table 2. Civil Penalty Actions During FY 1985
(continued)

Licensee	*Amount*	*Reason*
Princeton University Princeton, NJ EA 85-70	$4,000 proposed in FY 85; Pending	Violations included an individual receiving a skin exposure of 38 rems.
Toledo Edison Co. (Davis-Besse) EA 85-71	$100,000 proposed and paid in FY 85	Violations involved removal of a security-fire/radiation computer from service without proper notification, failure to monitor pipe leakage and failure to maintain reactor power for the indicated reactor coolant flow rate.
Syncor International Sylmar, CA EA 85-78	$2,500 proposed and paid in FY 85	Violation involved licensed material being left unattended and unsecured in unlocked vehicles with the keys in the ignition and the motor running.
Florida Power & Light Co. (Turkey Point) EA 85-80	$100,000 proposed in FY 85; Pending	Violation involved the failure of the licensee's staff to determine whether a modification to the spent fuel pits piping created an unreviewed safety question.
Astrotech, Inc. Harrisburg, PA EA 85-86	$5,000 proposed in FY 85; Pending	Violations involved inadequate management oversight and control of licensed facilities.
Hurley Medical Center Flint, MI EA 85-89	$2,500 proposed in FY 85; Pending	Violations involved a breakdown in management oversight and control of the licensee's health physics program.
Met-Chem Engineering Laboratories Salt Lake City, UT EA 85-92	$5,000 pending in FY 85; Pending	Violations involved the use of an unauthorized and unqualified individual to perform licensed activities and an overexposure to a radiographer.
Calumet Testing Services, Inc. Griffith, IN EA 85-93	$16,000 proposed in FY 85; Pending	Violation involved an overexposure greater than 75 rems to the hand of a radiographer.
American Electric Power Service Corp. (DC Cook) EA 85-94	$100,000 proposed in FY 85; Pending	Violation involved a failure to maintain adequate control over access to vital areas and a reporting violation.
Commonwealth Edison Co. (LaSalle) EA 85-95	$125,000 proposed in FY 85; Pending	Violations involved failure to ensure that modifications performed on safety-related systems were adequately controlled so that the operability of the systems was not jeopardized.
Kay-Ray, Inc. Arlington Heights, IL EA 85-96	$500 proposed and paid in FY 85	Violation involved an extremity exposure of 19.61 rems to one licensee employee in excess of the 18.75 rems regulatory limit.
Metro Health Center Erie, PA EA 85-98	$3,750 proposed in FY 85; Pending	Violations involved a breakdown in management oversight and control of the radiation safety program.
Commonwealth Edison Co. (Byron) EA 85-99	$50,000 proposed in FY 85; Pending	Violations involved the failure of management to follow radiation protection procedures, evaluate radiological conditions, or prepare adequate procedures.
Baltimore Gas & Electric (Calvert Cliffs) EA 85-102	$50,000 proposed in FY 85; Pending	Violation involved the licensee's failure to implement and maintain the installed upgraded post-accident sampling system.
Sacramento Municipal Utility District (Rancho Seco) EA 85-103	$50,000 proposed in FY 85; Pending	Violation involved a non-isolable primary coolant system leak.

Table 3. IE Orders Issued During FY 1985

Licensee	*Date*	*Reason*
Nuclear Pharmacy, Inc. Albuquerque, NM EA 84-100	October 25, 1984	Order Modifying Licenses (Effective Immediately) **Reason:** Licensee's ineffective and belated investigation of a molybdenum-99 breakthrough incident and the pervasiveness of its record-keeping problems.
Inspection and Testing, Inc. Chubbuck, ID EA 84-18	November 27, 1984	Order Revoking License (Effective Immediately) **Reason:** Licensee's failure to answer an Order to Show Cause issued August 31, 1984.
Boston Edison Company M/C Nuclear (Pilgrim Nuclear Power Station) EA 84-112	November 29, 1984	Order Modifying Licensee **Reason:** Licensee's failure to adequately plan, supervise and control activities involving the potential for personnel exposure to radiation in excess of regulatory limits.
Connecticut Yankee Atomic Power Co. (Haddam Neck Plant) EA 84-115	December 13, 1984	Order Modifying License **Reason:** Licensee's failure to adequately plan, direct, and control activities involving design modifications that have the potential for affecting the public health and safety.
Veterans Administration Richard L. Roudebush Veterans Administrative Medical Center Indianapolis, IN EA 84-10	December 24, 1984	Order Rescinding Order Imposing Civil Monetary Penalties **Reason:** NRC review of circumstances resulted in the conclusion that the civil penalties should be rescinded.
Veterans Administration Hospital Boston, MA EA 84-114	December 26, 1984	Order Modifying License **Reason:** Licensee's inadequate control of the radiation safety program demonstrated the need for significant corrective measures to prevent similar violations in the future.
Community Hospital Torrington, WY EA 84-124	December 31, 1984	Order Confirming Suspension of Use of Licensed Material **Reason:** Licensee's lack of adequate control resulted in numerous violations of NRC requirements and showed a lack of understanding and consequent disregard for the Commission's safety requirements.
Applied Health Physics, Inc. Bethel Park, PA EA 84-131	January 3, 1985	Order Modifying License Effective Immediately **Reason:** Violations which demonstrated that adequate control was not exercised over a shipment of radioactive waste.
Gorsira X-Ray, Inc. Farmington Hills, MI EA 85-02	January 15, 1985	Order to Show Cause and Order Suspending License Effective Immediately **Reason:** Licensee's failure to demonstrate sufficient financial resources as well as the ability and willingness to comply with NRC requirements.
Veterans Administration Medical Center Bronx, NY EA 84-98	March 5, 1985	Order Modifying License **Reason:** Exposure of a researcher.
Veterans Administration Medical Center Washington, DC EA 85-31	March 27, 1985	Order Modifying License **Reason:** Licensee's lack of followup to correct previously identified deficiencies.
Gorsira X-Ray, Inc. Farmington Hills, MI EA 85-02	April 2, 1985	Order Revoking License **Reason:** Licensee did not file an answer to the Order issued on January 15, 1985.

Table 3. IE Orders Issued During FY 1985
(continued)

Licensee	*Date*	*Reason*
John C. Haynes Co. Newark, OH EA 85-40	April 5, 1985	Order **Reason:** Licensee's willful disregard of the Commission's requirements, and the lack of adequate control over licensed activities.
Met Lab, Inc. Hampton, VA EA 85-04	May 15, 1985	Order to Show Cause Why License Should Not Be Revoked **Reason:** Licensee's willful submission of a material false statement.
Pittsburgh Testing Laboratory Pittsburgh, PA EA 85-57	May 24, 1985	Order to Show Cause Why License Should Not Be Suspended and Modified (Immediately Effective) **Reason:** Licensee deliberately assigned uncertified individuals to perform radiographic operations.
Tennessee Valley Authority (Sequoyah and Browns Ferry) EA 85-49	June 14, 1985	Order Modifying Licenses **Reason:** Concerns over the licensee's commitment to properly evaluate potentially significant safety conditions.
Advanced Medical Systems, Inc. Geneva, OH EA 85-60	June 28, 1985	Order Modifying License **Reason:** Apparent overexposure to a worker.
The Christ Hospital Cincinnati, OH EA 85-84	September 11, 1985	Confirmatory Order Modifying License **Reason:** Lack of formalized procedures which led to a misadministration.

Staff experience with the Readiness Review Programs to date indicate that the Readiness Review constitutes a promising approach to providing additional assurance to licensee management and to the NRC staff that a plant has been designed and constructed in accordance with licensing commitments and regulatory requirements. The NRC's incremental acceptance of completed work is expected to be a significant improvement over the prior practice of deferring major decisions until the final stages of construction.

QA Inspection Procedures. The NRC QA Report to Congress concluded that NRC QA inspection efforts have in the past focused too much on form and paperwork instead of confirming implementation of QA programs and the quality of the completed work. The report recommended the NRC QA inspection program be reoriented to give proper emphasis to QA program performance and effectiveness. The staff is revising the NRC QA inspection program for operating reactors accordingly. Revised QA inspection procedures that emphasize program implementation and QA program effectiveness are being developed, field tested, and incorporated into the NRC inspection program.

QA Standards Development

A regulation entitled "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Appendix B of 10 CFR Part 50) sets forth the quality assurance requirements for the design, construction, and operation of those structures, systems, and components having a safety-related function. Current guidance as to the controls that the NRC staff considers in compliance with those criteria is provided in a number of regulatory guides, endorsing industry standards. Efforts are under way to revise, consolidate where possible, or to develop new NRC Regulatory Guides addressing quality assurance program requirements for nuclear power plant design, construction, and operation. This effort is intended to reflect industry changes, the lessons described in the QA Report to Congress, recent events, and NRC inspections. The revised Regulatory Guide for QA for design and construction was issued in fiscal year 1985. Revisions to the Regulatory Guide for QA Operations are expected to be completed in fiscal year 1986 and issued in fiscal year 1987. Other similar activities which incorporate lessons learned will be undertaken after completion of these two major activities.

QA for Waste Management Activities

Federal regulations (10 CFR Part 60) require the U.S. Department of Energy (DOE) to implement a quality assurance (QA) program to provide confidence in the data and information necessary for obtaining an NRC authorization to construct—and eventually a license to operate—a permanent geological repository for high-level nuclear waste. In the next few years, DOE's site characterization activities will develop much of the technical data to support the DOE application for an NRC license. The NRC's Office of Nuclear Material Safety and Safeguards (NMSS) and IE are working jointly to develop guidance on quality assurance programs for site characterization activities to supplement the NRC regulatory requirements. A joint NMSS-IE task force is developing supplementary guidance to cover such specific waste management issues as research and exploration, peer review, QA for existing data, configuration management for conceptual design, QA for computer software, independence of the QA organization, and determination of the set of equipment and activities to which the QA program applies. The guidance will take the form of Generic Technical Positions, which for repository activities are similar to Regulatory Guides for reactor activities.

IE is also providing QA guidance or consultation to the other program offices of the NRC on such issues as decommissioning, independent spent fuel installations, and monitored retrievable storage facilities.

Design Assurance Activities

Integrated Design Inspection Program. As part of the program to improve the assurance of design quality and its implementation at nuclear power plants, the NRC has developed and implemented a program of Integrated Design Inspections (IDIs). An IDI provides a comprehensive examination of design development and design implementation for a selected safety-related system on a given reactor project. It encompasses the total design process from formulation of principal design and architectural criteria through the development and translation of the design and its revisions into the as-built configuration. The program includes inspection at licensee and architect-engineer offices as well as on-site verification of the design.

The results of the IDI are conveyed to the appropriate Regional and Headquarters Offices and are used as part of the overall NRC assessment of the plant prior to a decision on issuance of an operating license.

During fiscal year 1985, IDIs were performed for the Perry (Ohio) and Shearon Harris (N.C.) nuclear power plants. Follow-up inspections to examine corrective actions and close-out open issues were conducted for these plants and also for the River Bend (La.) and Seabrook (N.H.) nuclear power plants which had IDIs prior to fiscal year 1985. Although several design and hardware changes resulted from each of these inspections, the NRC was able to conclude that the design processes used for the various plants were adequately controlled. Some of the corrective actions taken by the design organizations were directed towards across-the-board actions to correct systematic problems identified during the IDI.

Specific examples of IDI findings include: (1) cabling for DC motor operated valves was found to be inadequately sized at the Perry plant, resulting in replacement of certain cables to ensure adequate voltage; (2) an Engineering Assurance review conducted as a result of the IDI for River Bend to assess weaknesses in design verification identified a violation of the single failure criteria for containment isolation in ECCS suction lines. A similar design problem was found for the Nine Mile Point Unit 2 (N.Y.); and (3) overload protection for motor operated valves at Shearon Harris had not been properly selected to protect the motors, resulting in replacement of the thermal overload devices.

Independent Design Verification Program. The Independent Design Verification Program (IDVP) has been addressed to the same purpose as the IDI, except that an independent contractor is used, either in lieu of or in addition to, direct NRC inspection. The IDVP involves a review of the design process, including a sample of design details, performed by an independent contractor hired by the applicant. The IDVP also includes elements of on-site verification in selected areas. Since IE assumed responsibility for the IDVP program in 1984, IDVPs have been performed at Byron (Ill.), Limerick (Pa.), Clinton (Ill.), and Hope Creek (N.J.).

During fiscal year 1985, IDVPs were completed for both Limerick and Clinton. For these two plants all issues developed by the IDVPs were resolved. The Byron IDVP, which was a consequence of the design issues raised by the IDI conducted by the NRC for Byron, was also completed. The Byron IDVP along with the staff's affidavit, provided the basis for the Licensing Board's action on design deficiencies.

Engineering Assurance Programs. Self-directed Engineering Assurance Programs (EAP) are also used to provide additional confidence in the adequacy of design and in the design control process. This program is similar to an IDVP in that the technical reviewers of the design and design process had no part in the original design work. However, an EAP differs from an IDVP in that the review may be conducted as an ongoing activity during the design and construction rather than taking place at or near the end of design and construction. It also differs in that these reviews are performed by technical personnel employed by the applicant as principal plant architect-engineer, rather than by independent contractor personnel. Plants in this category include South Texas Project, Millstone 3 (Conn.), and Nine Mile Point 2 (N.Y.).

These three EAP programs are currently in progress. To date, the South Texas Project EAP has identified three concerns that have been reported to the NRC (in accordance with the reporting requirement of 10 CFR 50.55(e)). In addition, design reviews are being integrated into the Vogtle and WNP-3 readiness reviews.

Although the above design assurance activities have resulted in a number of design and hardware changes, they have, on balance, confirmed that adequate design control processes are in place and, generally were properly implemented at the facilities reviewed. Overall, the design assurance activities have provided the additional assurance sought by NRC to verify that licensing commitments have been properly incorporated into the plant design.

GENERIC COMMUNICATIONS

The Office of Inspection and Enforcement (IE) informs licensees and construction permit holders of significant events, inspection findings, defects, and other matters of generic interest. Bulletins (which require action and response) and information notices (which simply transmit information) usually result from reports of events or defects, or from findings of Regional Offices or other offices.

Information Notices

The NRC Office of Inspection and Enforcement issues information notices to licensees and construction permit holders to inform them of events or other matters that may have generic implications. Many of these issuances are based on events reported by licensees, NRC Inspectors, Agreement States, or others, when a preliminary evaluation indicates that the event may be of interest to other licensees. A total of 109 NRC information notices were issued in fiscal year 1985, including 10 updates of previously issued information notices. (Table 4 lists all information notices issued in fiscal year 1985.) Information notices provide information but do not require specific actions. They are rapid transmittals of information which may not yet have been completely analyzed by the NRC, but of which licensees should be aware. Licensees receiving an Information Notice are expected to review the information for applicability to their facilities, and consider actions, if appropriate, to preclude a similar problem occurring at their facilities. The NRC then may follow up through inspections to ensure that licensees have an adequate system for reviewing and appropriately acting on information notices. Further, regional inspections are made on selected safety issues, including those addressed by some information notices.

Bulletins

Inspection and Enforcement also issues bulletins, which provide information about one or more similar events of significance and require that licensees take specific actions. The licensee reports back on actions taken or to be taken and provides information the NRC may need to assess the need for further action. Prompt response by licensees is required; failure to respond normally will result in NRC enforcement action. Before issuing a bulletin, the NRC may seek comments from

The NRC's Integrated Design Inspection (IDI) Program provides for careful examination of design development and implementation of safety-related systems at specific nuclear plants. Results of such inspections are sent to the NRC Regional Offices as well as NRC Headquarters for use in an overall assessment of the plant prior to issuance of an operating license. The Shearon Harris nuclear plant at New Hill, N.C., a 900-megawatt pressurized water reactor facility, shown above, was one of several installations for which IDIs or follow-up inspections were performed in 1985.

the nuclear industry. This technique has proven effective in generating faster and more informed responses from affected licensees. However, the nature of the problem and a need for timely action may limit such prior consultation. Bulletins generally require one-time action and are not intended as substitutes for formally issued regulations or for imposed license amendments.

No IE bulletins were issued during fiscal year 1985. However, just before the end of the fiscal year, a proposed bulletin on vapor binding of auxiliary feedwater pumps was presented to the Committee to Review Generic Requirements (CRGR), which recommended that the bulletin be issued. This will be accomplished early in fiscal year 1986.

Since 1983, IE has performed a formal closeout of selected bulletins that were issued during 1978 or more recently. The objectives of this effort are:

(1) To determine whether further generic actions are needed to fully resolve the issue. (Examples of possible actions are revisions to the inspection program, modification of licensing basis or technical specifications, and development of new regulatory guides.)

(2) To determine whether plant-specific regional action is needed.

Since the program was initiated, a total of 21 bulletins have been closed out, 10 during fiscal year 1985. (See Table 5.) Of the 21 bulletins closed out, five have resulted in recommendations for generic actions and five others in some general recommendations and conclusions. Almost all of the closeout efforts have resulted in plant-specific regional follow-up at some plants.

REPORTS OF DEFECTS

During fiscal year 1985, the staff continued to track notifications made to the NRC in accordance with NRC regulations, 10 CFR Parts 21 and 50.55(e). Part 21 requires vendors who supply components for nuclear power plants to report any defects or noncompliance with NRC requirements that could create a substantial safety hazard. Part 50.55(e) requires holders of construction permits (CP) for nuclear power plants to make similar reports. These requirements result in several thousand reports to the NRC each year. The Office of Inspection and Enforcement in conjunction with the Regional Offices reviews the reports to determine; 1) whether corrective action taken by the vendor or CP-holder is sufficient, 2) whether an information notice or bulletin should be issued, and 3) whether other actions may be required, such as an inspection, a temporary instruction to the Regional Offices, or further correspondence with the vendor or CP-holder. Several fiscal year 1985 Information Notices resulted from Part 21 and 50.55(e) reports.

A computer-based system tracks the actions taken by vendors, CP-holders, and the NRC; allows automated searches of information on file; and assists in communicating the information between NRC Headquarters and the Regional Offices. The processing of Part 21 reports by IE and the Regional Offices was examined during fiscal year 1985. Improvements made as a result of this review include: (1) periodic distribution to Regional Offices and NRR of a comprehensive list of Part 21 reports; (2) direct distribution of all Part 21 reports to Regional Offices; and (3) more effective tracking and closeout of generic issues raised by Part 21 reports. Also, analysis of Part 21 reports by IE for a small number of selected plants confirmed the effectiveness of Part 21 in notifying the NRC of generic safety issues.

INCIDENT RESPONSE

Events Analysis

The Nuclear Regulatory Commission maintains a 24-hour-a-day, 365-day-a-year Operations Center. Located in Bethesda, Md., the Operations Center is the NRC's point of direct communication through dedicated telephone lines for reports of significant events at licensed nuclear power plants and certain fuel cycle facilities.

The staff at the Operations Center evaluates telephone notifications and, depending on the safety significance of the event, notifies appropriate NRC headquarters personnel and other Federal agencies. In all cases, the NRC Regional Office responsible for the facility reporting the event is notified. Response to an event may vary from simply recording the circumstances of the event for later evaluation to immediately activating response organizations within Headquarters and the affected NRC Region. Upon activation, these response organizations monitor the event to ensure that appropriate actions are being taken to protect the health and safety of the public. The NRC recognizes that early in a rapidly moving severe accident sequence, the agency's role is secondary to that of the licensee and off-site organizations whose immediate response has been defined ahead of time in their emergency planning.

Each event reported to the Operations Center by a licensee or Regional Office is evaluated to determine whether there are any generic implications for other facilities. Event reports are screened early during the first working day after receipt. Events that may be significant from a generic standpoint receive additional in-depth evaluation. For events found to have significant generic implications, the NRC issues an Information Notice or a Bulletin to appropriate licensees and construction permit holders. Some examples of events evaluated because of their generic implications are discussed below.

Monticello. On December 4, 1984, the Monticello plant in Minnesota experienced excessive scram times during surveillance testing of the control rod system, following an extended outage to replace major portions of the recirculation system piping. The excessive scram times were caused by clogged inner filters in the control rod drive mechanisms. The filters were clogged by fibers from soluble dams used to contain an inert gas for welding during the pipe replacement. An Information Notice (IN 85-13) was issued to inform licensees

Table 4. IE Information Notices Issued in FY 1985

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
84-75	Calibration Problems—Eberline Instrument Model 6112B Analog Teletectors	10/05/84	All power reactor facilities holding an OL or CP; research and test reactors; and fuel facilities
84-76	Loss of All AC Power	10/19/84	All power reactor facilities holding an OL or CP
84-77	Incident Involving Teletherapy Unit (AECL ELDORADO-78)	10/26/84	All teletherapy licensees authorized to possess AECL cobalt-60 teletherapy units
84-78	Underrated Terminal Blocks That May Adversely Affect Operation of Essential Electrical Equipment	11/2/84	All power reactor facilities holding an OL or CP
84-79	Failure to Properly Install Steam Separator at Vermont Yankee	11/5/84	All BWR facilities holding an OL or CP
84-80	Plant Transients Induced by Failure of Non-Nuclear Instrumentation Power	11/8/84	All B&W power reactor facilities holding an OL or CP
84-81	Inadvertent Reduction in Primary Coolant Inventory in Boiling Water Reactors During Shutdown and Startup	11/16/84	All BWR facilities holding an OL or CP
84-48 Supp 1	Failures of Rockwell International Globe Valves	11/16/84	All power reactor facilities holding an OL or CP
84-82	Guidance for Posting Radiation Areas	11/19/84	All power reactor facilities holding an OL or CP
84-83	Various Battery Problems	11/19/84	All power reactor facilities holding an OL or CP
84-84	Deficiencies in Ferro-resonant Transformers	11/27/84	All power reactor facilities holding an OL or CP
84-85	Molybdenum Breakthrough from Technetium-99m Generators	11/30/84	All NRC medical licensees and Radio-pharmaceutical suppliers
84-86	Isolation Between Signals of the Protection System and Non-Safety Related Equipment	11/30/84	All power reactor facilities holding an OL or CP
84-87	Piping Thermal Deflection Induced by Stratified Flow	12/3/84	All power reactor facilities holding an OL or CP
84-88	Standby Gas Treatment System Problems	12/3/84	All BWR facilities holding an OL or CP
84-89	Stress Corrosion Cracking in Nonsensitized 316 Stainless Steel	12/7/84	All BWR facilities holding an OL or CP
84-90	Main Steam Line Break on Environmental Qualification of Equipment	12/7/84	All PWR and gas cooled power plants holding an OL or CP
84-91	Quality Control Problems of Meteorological Measurements Programs	12/10/84	All power reactor facilities holding an OL or CP
84-92	Cracking of Flywheels on Cummins Fire Pump Diesel Engines	12/17/84	All power reactor facilities holding an OL or CP; research and test reactors; and fuel facilities
84-93	Potential for Loss of Water from the Refueling Cavity	12/17/84	All power reactor facilities holding an OL or CP except Fort St. Vrain

Table 4. IE Information Notices Issued in FY 1985
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
84-94	Reconcentration of Radionuclides Involving Discharges into Sanitary Sewage Systems Permitted Under 10 CFR 20.303	12/21/84	All NRC materials licensees other than licensees that use sealed sources only
85-01	Continuous Supervision of Irradiators	1/10/85	All materials licensees possessing irradiators that are not self-shielded & contain more than 10,000 curies of radioactive material
85-02	Improper Installation And Testing Of Differential Pressure Transmitters	1/15/85	All power reactor facilities holding an OL or CP
85-03	Separation Of Primary Reactor Coolant Pump Shaft And Impeller	1/15/85	All PWR facilities holding an OL or CP
85-04	Inadequate Management Of Security Response Drills	1/17/85	All power reactor facilities holding an OL or CP; and fuel fabrication and processing facilities using or processing a formula quantity of special nuclear material
85-05	Pipe Whip Restraints	1/23/85	All power reactor facilities holding an OL or CP
85-06	Contamination Of Breathing Air Systems	1/23/85	All power reactor facilities holding an OL or CP
85-07	Contaminated Radiography Source Shipments	1/29/85	All NRC licensees authorized to possess industrial radiography sources
85-08	Industry Experience On Certain Materials Used In Safety-Related Equipment	1/30/85	All power reactor facilities holding an OL or CP
85-09	Isolation Transfer Switches And Post-Fire Shutdown Capability	1/31/85	All power reactor facilities holding an OL or CP
85-10	Post-tensioned Containment Tendon Anchor Head Failure	2/6/85	All power reactor facilities holding an OL or CP
85-11	Licensee Programs For Inspection Of Electrical Raceway And Cable Installations	2/11/85	All power reactor facilities holding an OL or CP
85-12	Recent Fuel Handling Events	2/11/85	All power reactor facilities holding an OL or CP
85-13	Consequences Of Using Soluble Dams	2/21/85	All BWR and PWR reactor facilities holding an OL or CP
85-14	Failure Of A Heavy Control Rod (B4C) Drive Assembly To Insert On A Trip Signal	2/22/85	All power reactor facilities holding an OL or CP
85-15	Nonconforming Structural Steel For Safety-Related Use	2/22/85	All power reactor facilities holding an OL or CP
85-16	Time/Current Trip Curve Discrepancy Of ITE/Siemens-Allis Molded Case Circuit Breaker	2/27/85	All power reactor facilities holding an OL or CP
85-17	Possible Sticking of ASCO Solenoid Valves	3/1/85	All power reactor facilities holding an OL or CP
83-70 Supp 1	Vibration-Induced Valve Failures	3/4/85	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1985
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
85-18	Failures Of Undervoltage Output Circuit Boards In The Westinghouse-Designed Solid State Protection System	3/7/85	All Westinghouse PWR power reactor facilities holding an OL or CP
85-10 Supp 1	Post-tensioned Containment Tendon Anchor Head Failure	3/8/85	All power reactor facilities holding an OL or CP
85-19	Alleged Falsification Of Certifications And Alteration Of Markings On Piping, Valves, And Fittings	3/11/85	All power reactor facilities holding an OL or CP
85-20	Motor-Operated Valve Failures Due To Hammering Effect	3/12/85	All power reactor facilities holding an OL or CP
85-21	Main Steam Isolation Valve Closure Logic	3/18/85	All PWR facilities holding an OL or CP
85-22	Failure Of Limitorque Motor-Operated Valves Resulting From Incorrect Installation Of Pinion Gear	3/21/85	All power reactor facilities holding an OL or CP
85-23	Inadequate Surveillance And Postmaintenance And Post-modification System Testing	3/22/85	All power reactor facilities holding an OL or CP
85-24	Failures Of Protective Coatings In Pipes And Heat Exchangers	3/26/85	All power reactor facilities holding an OL or CP
85-25	Consideration Of Thermal Conditions In The Design And Installation Of Supports For Diesel Generator Exhaust Silencers	4/2/85	All power reactor facilities holding an OL or CP
85-26	Vacuum Relief System For Boiling Water Reactor Mark I And Mark II Containments	4/2/85	All BWR facilities having a Mark I or Mark II containment or holding an OL or CP
85-27	Notifications To The NRC Operations Center And Reporting Events In Licensee Event Reports	4/3/85	All power reactor facilities holding an OL or CP
85-28	Partial Loss Of AC Power And Diesel Generator Degradation	4/9/85	All power reactor facilities holding an OL or CP
85-03 Supp 1	Separation Of Primary Reactor Coolant Pump Shaft And Impeller	4/9/85	All PWR facilities holding an OL or CP
85-29	Use Of Unqualified Sources In Well Logging Applications	4/12/85	All well logging source licensees
85-30	Microbiologically Induced Corrosion Of Containment Service Water System	4/19/85	All power reactor facilities holding an OL or CP
85-31	Build-up Of Enriched Uranium In Ventilation Ducts And Associated Effluent Treatment Systems	4/19/85	All uranium fuel fabrication licensees
85-32	Recent Engine Failures Of Emergency Diesel Generators	4/22/85	All power reactor facilities holding an OL or CP
85-33	Undersized Nozzle-To-Shell Welded Joints In Tanks And Heat Exchangers Constructed Under The Rules Of The ASME Boiler And Pressure Vessel Code	4/22/85	All power reactor facilities holding an OL or CP
84-84 Rev 1	Deficiencies In Ferro-Resonant Transformers	4/24/85	All power reactor facilities holding an OL or CP
85-34	Heat Tracing Contributes To Corrosion Failure Of Stainless Steel Piping	4/30/85	All power reactor facilities holding an OL or CP
85-35	Failure Of Air Check Valves To Seat	4/30/85	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1985
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
84-52 Supp 1	Inadequate Material Procurement Controls On The Part Of Licensees And Vendors	5/8/85	All power reactor facilities holding an OL or CP
85-36	Malfunction Of A Dry-Storage, Panoramic, Gamma Exposure Irradiator	5/9/85	All licensees possessing gamma irradiators
85-20 Supp 1	Motor-Operated Valve Failures Due To Hammering Effect	5/14/85	All power reactor facilities holding an OL or CP
84-55 Supp 1	Seal Table Leaks At PWRs	5/14/85	All power reactor facilities holding an OL or CP
85-37	Chemical Cleaning Of Steam Generators At Millstone 2	5/14/85	All PWR facilities holding an OL or CP
85-38	Loose Parts Obstruct Control Rod Drive Mechanism	5/21/85	All PWR facilities designed by B&W holding an OL or CP
85-39	Auditability Of Electrical Equipment Qualification Records At Licensees' Facilities	5/22/85	All power reactor facilities holding an OL or CP
85-40	Deficiencies In Equipment Qualification Testing And Certification Process	5/22/85	All power reactor facilities holding an OL or CP
85-41	Scheduling Of Pre-Licensing Emergency Preparedness Exercises	5/24/85	All power reactor facilities holding an OL or CP
85-42	Loose Phosphor In Panasonic 800 Series Badge Thermoluminescent Dosimeter (TLD) Elements	5/29/85	All power reactor facilities holding an OL or CP
85-43	Radiography Events At Power Reactors	5/30/85	All power reactor facilities holding an OL or CP
85-44	Emergency Communication System Monthly Test	5/30/85	All power reactor facilities holding an OL
85-45	Potential Seismic Interaction Involving The Movable In-Core Flux Mapping System Used In Westinghouse Designed Plants	6/6/85	All power reactor facilities holding an OL or CP
85-46	Clarification Of Several Aspects Of Removable Radioactive Surface Contamination Limits For Transport Packages	6/10/85	All power reactor facilities holding an OL
85-47	Potential Effect Of Line-Induced Vibration On Certain Target Rock Solenoid-Operated Valves	6/18/85	All power reactor facilities holding an OL or CP
85-48	Respirator Users Notice: Defective Self-Contained Breathing Apparatus Air Cylinders	6/19/85	All power reactor facilities holding an OL or CP; research and test reactors; fuel cycle and Priority 1 material licensees
85-49	Relay Calibration Problem	7/1/85	All power reactor facilities holding an OL or CP
85-50	Complete Loss Of Main And Auxiliary Feedwater At A PWR Designed By Babcock & Wilcox	7/8/85	All power reactor facilities holding an OL or CP
85-51	Inadvertent Loss Or Improper Actuation Of Safety-Related Equipment	7/10/85	All power reactor facilities holding an OL or CP
85-52	Errors In Dose Assessment Computer Codes And Reporting Requirements Under 10 CFR Part 21	7/10/85	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1985
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
85-53	Performance Of NRC-Licensed Individuals While On Duty	7/12/85	All power reactor facilities holding an OL or CP
85-54	Teletherapy Unit Malfunction	7/15/85	All NRC licensees authorized to use teletherapy units
85-55	Revised Emergency Exercise Frequency Rule	7/15/85	All power reactor facilities holding an OL or CP
85-56	Inadequate Environment Control For Components And Systems In Extended Storage Or Layup	7/15/85	All power reactor facilities holding an OL or CP
85-57	Lost Iridium-192 Source Resulting In The Death Of Eight Persons In Morocco	7/16/85	All power reactor facilities holding an OL or CP; fuel facilities; and material licensees
85-58	Failure Of A General Electric Type AK-2-25 Reactor Trip Breaker	7/17/85	All power reactor facilities designed by B&W and CE holding an OL or CP
85-59	Valve Stem Corrosion Failures	7/17/85	All power reactor facilities holding an OL or CP
85-60	Defective Negative-Pressure, Air-Purifying, Full Facepiece Respirators	7/17/85	All power reactor facilities holding an OL or CP; research and test reactors; fuel facilities; and material licensees
85-61	Misadministrations To Patients Undergoing Thyroid Scans	7/22/85	All power reactor facilities holding an OL and certain fuel facilities
85-62	Backup Telephone Numbers To The NRC Operations Center	7/23/85	All power reactor facilities holding an OL and certain fuel facilities
85-63	Potential For Common-Mode Failure Of Standby Gas Treatment System On Loss Of Off-Site Power	7/25/85	All power reactor facilities holding an OL or CP
85-64	BBC Brown Boveri Low-Voltage K-Line Circuit Breakers, With Deficient Overcurrent Trip Devices Models OD-4 And 5	7/26/85	All power reactor facilities holding an OL or CP
85-65	Crack Growth In Steam Generator Girth Welds	7/31/85	All PWR facilities holding an OL or CP
85-66	Discrepancies Between As-Built Construction Drawings And Equipment Installations	8/7/85	All power reactor facilities holding an OL or CP
85-67	Valve-Shaft-To-Actuator Key May Fall Out Of Place When Mounted Below Horizontal Axis	8/8/85	All power reactor facilities holding an OL or CP
85-42 Rev 1	Loose Phosphor In Panasonic 800 Series Badge Thermoluminescent Dosimeter (TLD) Elements	8/12/85	Materials and fuel cycle licensees
85-68	Diesel Generator Failure At Calvert Cliffs Nuclear Station Unit 1	8/14/85	All power reactor facilities holding an OL or CP
85-69	Recent Felony Conviction For Cheating On Reactor Operator Requalification Tests	8/15/85	All power reactor facilities holding an OL or CP
85-70	Teletherapy Unit Full Calibration and Qualified Expert Requirements (10 CFR 35.23 And 10 CFR 35.24)	8/15/85	All material licensees
85-71	Containment Integrated Leak Rate Tests	8/22/85	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1985
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
85-72	Uncontrolled Leakage Of Reactor Coolant Outside Containment	8/22/85	All BWRs holding an OL or CP
85-73	Emergency Diesel Generator Control Circuit Logic Design Error	8/23/85	All power reactor facilities holding an OL or CP
84-70 Supp 1	Reliance On Water Level Instrumentation With A Common Reference Leg	8/26/85	All power reactor facilities holding an OL or CP
85-74	Station Battery Problems	8/29/85	All power reactor facilities holding an OL or CP
85-75	Improperly Installed Instrumentation, Inadequate Quality Control, And Inadequate Postmodification Testing	8/30/85	All power reactor facilities holding an OL or CP
85-76	Recent Water Hammer Events	9/19/85	All power reactor facilities holding an OL or CP
85-77	Possible Loss Of Emergency Notification System Due To Loss of AC Power	9/20/85	All power reactor facilities holding an OL or CP
85-78	Event Notification	9/23/85	All power reactor facilities holding an OL or CP
85-79	Inadequate Communications Between Maintenance, Operations, And Security Personnel	9/30/85	All power reactor facilities holding an OL or CP; research and nonpower reactor facilities; and fuel fabrications and processing facilities

and applicants of the impact of foreign material in the reactor coolant system on BWR scram times, the potential for introducing insoluble fibers into the reactor coolant system when using soluble dams, and the importance of ensuring the cleanliness of reactor coolant system water following major maintenance.

Browns Ferry. On February 13, 1985, during a reactor startup at the Browns Ferry Nuclear Plant Unit 3 in Alabama, a half scram occurred on low-reactor water level. A few minutes before the half scram, the operators had noticed that two of the three narrow-range water level instruments were reading approximately 40 inches. The third narrow-range instrument was indicating approximately 10 inches. Two wide-range level instruments were also observed by the operators to be indicating approximately 40 inches. At the time of the half scram, reactor coolant temperature was approximately 286F. (The normal full power temperature is approximately 550F.) Although four of the level instruments observed by the operators indicated nearly normal reactor water level (33 +/- 5 inches), actual reactor water level was approximately 10 inches. The operators incorrectly concluded that the narrow-range instrument indicating 10 inches was erroneous, since the four other level instruments were indicating approximately 40 inches.

The two narrow range level instruments that indicated 40 inches share a common reference leg. This reference leg apparently had lost some of its water inventory, causing all level-indicating instrumentation that tapped off that leg to indicate an erroneously high level. The two wide-range level instruments each have separate reference columns not shared by any of the narrow-range instruments. However, at the reactor coolant temperature existing at the time of the event, the wide range instrument readings would have been greater than 60 inches if the actual reactor water level were a normal 33 +/- 5 inches, because the wide range instruments are calibrated to give correct readings with reactor coolant temperature at its normal value for operation at power. The operators did not check the shutdown vessel flooding range level indication, which is calibrated for cold plant conditions. This instrument would have confirmed actual low water level conditions. The plant operators did not recognize the level instruments problem until some time after the half scram occurred. In fact, the licensee continued the reactor startup. A supplement to Information Notice 84-70, "Reliance on Water Level Instrumentation With a Common Reference Leg," was issued to reemphasize the need for operators to be cognizant of level instruments that share a common reference leg and to recognize that a problem in the reference leg will simultaneously affect all instruments that share that reference leg. The supplement

Table 5. IE Bulletins Closed Out in FY 1985

Bulletin No.	*Subject*
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems
79-12	Short-Period Scrams at Boiling Water Reactors
79-25	Westinghouse BFD Relays
80-07	BWR Jet Pump Assembly
80-12	Decay Heat Removal System Operability
80-25	Operating Problems with Target Rock Safety-Relief Valves at BWRs
81-01	Surveillance of Mechanical Snubbers
82-02	Degradation of Threaded Fasteners in Reactor Coolant Pressure Boundary of PWR Plants
84-01	Cracks in Boiling Water Reactor Mark I Containment Vent Headers

also discussed the need for operator training to emphasize the effects of system temperature and/or pressure on level instrument readings.

Rancho Seco. On June 1, 1985, an emergency diesel generator at the Rancho Seco Nuclear Power Generating Station in California was in the maintenance shutdown control mode—after being secured from an operational condition—when an emergency bus was deliberately de-energized for planned work on a parallel bus. The plant was shut down for refueling at the time. The de-energizing of the emergency bus created an undervoltage condition equivalent to a loss of off-site power on the bus. The diesel generator came up to design speed, but the diesel generator output breaker continuously cycled open and closed, thereby rendering the diesel generator inoperable. Investigation by the licensee indicated that the cycling of the output breaker was the result of a design error in the diesel generator control circuit logic. The design error was in the interface to the diesel generator control logic provided by the architect-engineer. An information notice (IN 85-73) was issued to inform licensees and applicants of the potentially significant emergency diesel generator control logic error.

Davis Besse. On June 9, 1985, the Davis-Besse plant in Ohio experienced a loss of main and auxiliary feedwater. At 1:35 a.m., one of the two main feedwater pumps tripped (stopped) on overspeed while the plant was operating at 90 percent power. Thirty seconds later, the reactor and turbine were automatically tripped on high reactor coolant system pressure. Seven seconds after the reactor tripped, both main steam isolation valves unaccountably closed, resulting in a loss of steam to the second main feedwater pump. At 1:40 a.m. steam generator levels began to fall from their normal post-trip level as the second main feedwater pump coasted down. Subsequently, all sources of feedwater to the steam generators were lost as the result of: an operator error; malfunctions of two redundant valves in the safety-related auxiliary feedwater system; and overspeed trips of the two redundant, steam turbine-driven auxiliary feedwater pumps.

Separate actions by operators were required (1) to correct the initial operator error, (2) to open the valves which malfunctioned, and (3) to reset the overspeed trips of the turbine-driven auxiliary feedwater pumps. Actions outside the control room were required to open the valves and place the pumps in operation. While operators acted to restart the safety-related auxiliary feedwater system, operator actions outside the control room were also taken to place an electric motor-driven, startup feedwater pump, that was not safety-related, in service.

The two steam generators were without feedwater for approximately 12 minutes; they had essentially boiled dry before feedwater, from any source, was made available. A number of other equipment problems complicated the event, but the operators were nonetheless successful in bringing the plant to a stable shutdown and in preventing any abnormal releases of radioactivity and any major plant damage.

The NRC Headquarters Operations Center was informed of the event by the licensee at 2:11 a.m., by which time the plant was in a stable, safe condition. In view of the equipment failures which had occurred, however, and the operator actions which had been necessary to regain auxiliary feedwater, Region III sent personnel to the site on the day of the event to evaluate the plant transient and its causes. On June 10, 1985, the day after, in conformance with the staff-proposed Incident Investigation Program (see Chapter 4), the NRC Executive Director for Operations sent an NRC team of technical experts to the site. The team of four staff members was instructed to: (1) establish the factual record as to what happened; (2) identify the probable cause as to why it happened; and (3) make appropriate findings and conclusions on which to base possible follow-on actions.

The team's investigation was completed on July 22, 1985, and the results were published in NUREG-1154, "Loss of Main

NRC Chairman Nunzio J. Palladino and members of the NRC senior staff are briefed during an exercise in the NRC Operations Center in Bethesda, Md. The Chairman and NRC Commissioners are frequent visitors to the Center.

and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985." The team concluded that the underlying cause of the multiple failures leading to loss of main and auxiliary feedwater was the licensee's lack of attention to detail in the care of plant equipment.

Information Notice 85-50, "Complete Loss of Main and Auxiliary Feedwater at a PWR Designed by Babcock and Wilcox," was issued on July 8, 1985, to inform industry of the Davis-Besse event and its possible generic implications. Further generic actions are being planned for the coming fiscal year.

Hurricane Gloria. On September 27, 1985, hurricane Gloria passed along the east coast of the United States from North Carolina to Long Island, eventually crossing Long Island, Connecticut, Massachusetts, New Hampshire, and Vermont. Brunswick 1 and 2 (N.C.), Oyster Creek 1 (N.J.), Limerick 1 (Pa.), Peach Bottom 2 and 3 (Pa.), Haddam Neck (Conn.), Pilgrim 1 (Mass.), Calvert Cliffs 1 and 2 (Md.), Millstone 1 and 2 (Conn.), Shoreham (N.Y.), Indian Point 2 and 3 (N.Y.), Maine Yankee, Surry 1 and 2 (Va.), and Salem 1 and 2 (N.J.) were near the path of this hurricane. Most of these plants were shut down before the hurricane arrived. Surry 1 and 2 and Pilgrim reduced power. Calvert Cliffs 1 and 2 and Salem 2 continued to operate at 100 percent power. The affected plants added additional personnel to operate their Technical Support Centers and Emergency Operations Facilities.

Because of the hurricane, Millstone Units 1 and 2 lost access to off-site power for approximately 24 hours; it was specifically due to switchyard arcing brought about by the by salt spray. The units had been shut down before losing access to off-site power. Salem Unit 1 eventually reduced power to 35 percent when the cooling water intake became clogged.

Additional staff were assigned by the Office of Inspection and Enforcement and the Office of Nuclear Reactor Regulation to the NRC Headquarters Operations Center to monitor the storm and evaluate licensee actions during the time the hurricane was passing near nuclear power plants. The additional staff included a response coordination team member, a second operations officer, a meteorologist, a hydrologist, various project managers, and a senior manager. Operations Center personnel maintained contact with Regions I and II as the hurricane traversed each region. Each region staffed its Incident Response Center to track the storm and to monitor the facilities at risk. At least two NRC personnel were at each site. The two regions provided periodic and thorough updates to the headquarters operations Center staff. Region-based incident investigations are augmented, when necessary, by headquarters technical experts. Such assistance was provided on six occasions during fiscal year 1985.

Operations Center

Construction of the new NRC Operations Center was completed by the end of January 1985. Final acceptance testing of the facility occurred in February, and the operational changeover occurred on February 27, 1985. March through August provided time to test the operational effectiveness of the Operations Center and to train response personnel in its use for a variety of exercise scenarios. Each of these exercises demonstrated operational improvements to assist the Commission in protecting the health and safety of the public. Among the improvements were: (1) quantity, quality, and dedication of spce, (2) a better telephone system, (3) sophisticated audio/video displays, and (4) a dedictated computer system.

As the fiscal year drew to a close, the Operations Center was involved in several real events which, while not requiring complete activation, did provide an opportunity to use the new facility. Hurricanes "Elena" and "Gloria" (discussed above) required close monitoring, the former as it threatened Crystal River (Fla.) and Waterford (La.) nuclear plants and the latter as it moved up the east coast threatening coastal licensees from Brunswick (N.C.) to Maine Yankee. An Alert at Dresden II (Ill.) caused the NRC to be placed in the Standby Mode of activation, with staffing of the Operations Center with a few appropriate technical experts. The NRC response to these events demonstrated significant improvements in the Commission's technical and administrative response capability.

Chief of Environmental Radiation for the Pennsylvania Department of Environmental Resources is Maggie Riley, shown describing actions of the State's Bureau of Radiation Protection during a "Relocation Tabletop Exercise," conducted at the National Emergency Training Center in Emmitsburg, Md., in December 1985. The exercise scenario postulated a radiation release from the Beaver Valley nuclear plant in Pennsylvania, and it involved more than 100 participants from 13 Federal and State agencies, as well as the American Red Cross and private nuclear insurance and utility organizations.

Regional Response Capability

The extent of regional office response to an incident is based on a pre-defined classification of events and associated NRC response modes. For a significant event, a regional base team and a regional site team are assembled. The base team monitors licensee performance and represents the NRC role until a decision is made to dispatch a team to the site. If the decision is made, the site team goes to the site and is responsible for coordinating the NRC's on-site incident response activities.

Regional response capabilities are assessed to ensure an adequate agency-wide response capability. The major evaluation in 1985 was of the Regions' interaction with other Federal agencies. This evaluation followed and was based on the successful Federal Field Exercise held in 1984 (see the 1984 NRC Annual Report, p. 121).

Emergency Response Procedures

Emphasis in fiscal year 1985 was placed on the development of detailed technical response procedures to be used in responding to transportation events. These procedures were successfully tested during an exercise with Region III, the State of Illinois, and IE Headquarters. Technical procedures developed during this period for safeguards events will be tested during the coming year.

Activation of the new Operations Center in February was followed by shakedown drills to test the application of existing and new procedures. These drills culminated in an exercise on April 10, 1985, which tested and confirmed the overall effectiveness of the new Operations Center.

During the report period, efforts were launched to develop an Emergency Response Data System (ERDS) for use during emergencies at commercial nuclear power plants. The ERDS concept provides for licensee-activated automatic transmission of pre-selected plant data from a licensee's existing emergency data computer to a computer at the NRC Operations Center.

The design phase of ERDS development has included: (1) surveys of existing electronic data systems at operating and nearly completed nuclear power plants, and (2) determination of hardware and software requirements at licensee facilities and at the NRC Operations Center.

Emergency Response Training

Development began during the year of a standard response training program and a formal training manual for NRC response personnel.

The NRC presented instruction on protective action decisionmaking and radiological assessment to State response personnel at the Federal Emergency Management Agency (FEMA) Training Center and to licensee personnel, at a course sponsored by the Institute of Nuclear Power Operations (INPO). The NRC also worked with FEMA to update their radiological assessment courses for State and local government personnel.

Federal Response Capability

The Federal Field Exercise in 1984 disclosed a need for Federal agencies to demonstrate their emergency response capabilities and support to State and local authorities. IE and regional staff participated in conferences which brought together more than 500 representatives of various groups which would respond to a nuclear power plant accident. The three sessions, which were held in Las Vegas, Atlanta, and Chicago, utilized plenary sessions and "hands-on" demonstrations to show what technical assistance the Federal government could provide.

Continuity of Government

During fiscal year 1985, the Commission reevaluated its role for responding to a national emergency. The Commission believes that it has certain essential functions to perform in a national emergency, but—contrary to certain earlier judgments—has decided that these functions are interruptible. The NRC, therefore, informed the Director of FEMA of its decision and requested that FEMA identify NRC as a "Category B" agency with respect to national emergencies.

EMERGENCY PREPAREDNESS

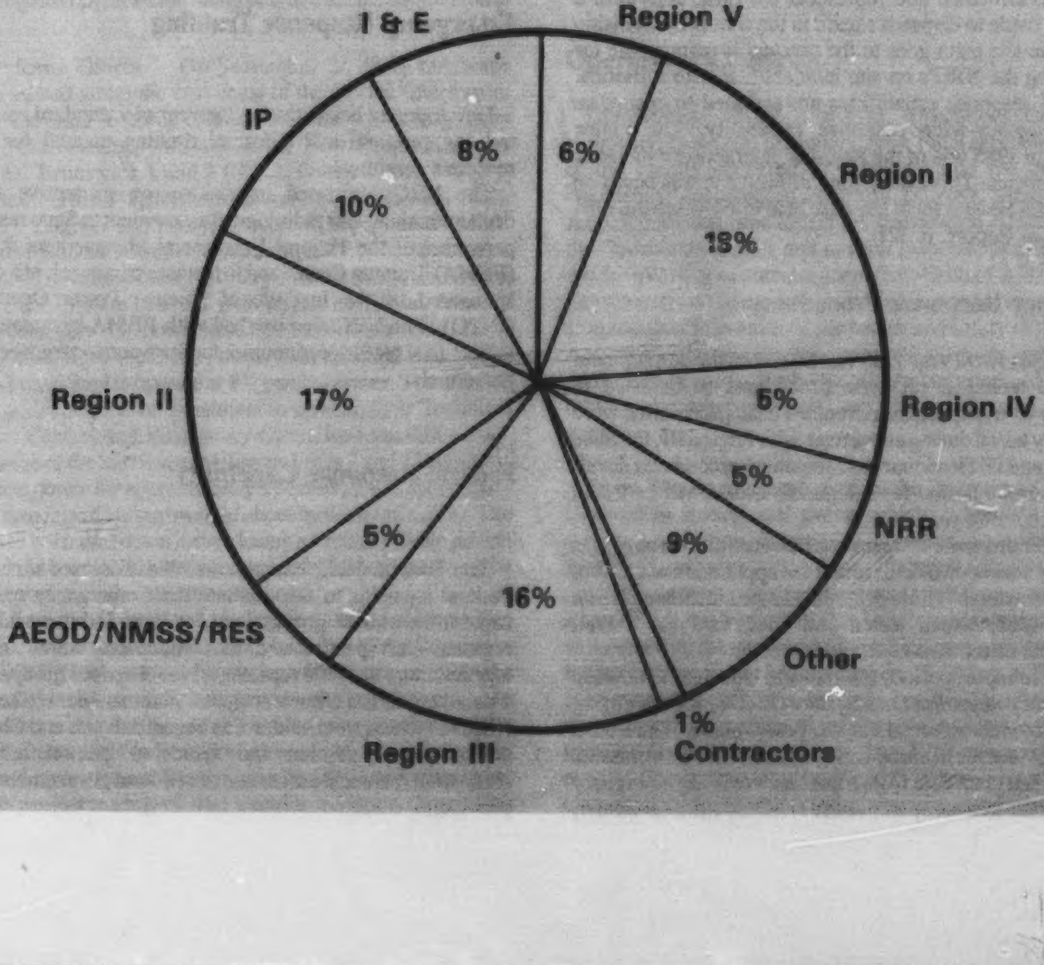
Support to Licensing Activities

During the report period, IE staff continued to evaluate the adequacy of applicant on-site plans to be included in the Safety Evaluation Report, and supplements thereto, for each plant in a near-term licensing status (designated NTOLs). The staff also took part in licensing hearings before Atomic Safety and Licensing Board panels and served on inspection teams appraising applicants' implementation of emergency preparedness programs and their full-participation exercises. NTOLs appraised during fiscal year 1985 included River Bend (La.), Perry (Ohio), Harris (N.C.), Millstone 3 (Conn.), and Hope Creek (N.J.). In addition, exercises were observed at nine NTOL sites. The staff also reviewed evaluations by the Federal Emergency Management Agency of off-site emergency plans for these facilities, as well as FEMA reports on State and local government performance during emergency preparedness exercises.

Emergency Response Facilities

During fiscal year 1985, appraisal of emergency response facilities (ERFs) continued. The adequacy of these support facilities for nuclear power plants is appraised against

FY 85 TECHNICAL TRAINING CENTER STUDENT PARTICIPATION



requirements of Supplement 1 to NUREG-0737, as issued in generic letter 82-33. ERF appraisals were completed at Arkansas, Catawba (S.C.), Diablo Canyon (Cal.), McGuire (N.C.), and WNP-2 (Wash.). Evaluations will extend over the next several years, as ERFs are completed at each plant.

Technical Training Program

NRC's Technical Training Center (TTC), located in Chattanooga, Tenn., has primary responsibility for the training of NRC employees in specialized technology areas related to regulation, inspection, and enforcement. The TTC currently offers 70 different highly specialized technical training courses designed to give NRC inspectors and other personnel the appropriate background to perform inspections and evaluations of commercial nuclear power plants, fuel fabrication and byproduct utilization facilities, test and research reactors and vendor facilities.

Although the courses were designed to provide specialized technical training to meet specific job requirements of NRC engineers, participants come from all NRC offices. Representatives of other government agencies, NRC contractors and foreign nationals may also attend when priorities permit.

In fiscal year 1985, the TTC presented or coordinated attendance at 104 courses. A total of 1,463 students attended and 1,734 student weeks of instruction were given. Two courses were presented through the Office of International Programs; a two-week BWR systems course was presented in Mexico in November 1984; and a two-week PWR systems course was presented in the People's Republic of China in April 1985. These courses were taught by members of the TTC staff with travel costs assumed by the International Atomic Energy Agency.

Training courses are conducted in conventional classrooms, scientific laboratories, nuclear power plants, and reactor control room simulators at the TTC and contractor locations throughout the United States.

Substantial efforts were made during fiscal year 1985 to improve the quality of training provided by the TTC. An engineering scale model of the Hartsville (Tenn.) BWR/6 reactor plant was leased from the Tennessee Valley Authority and moved to the TTC.

A lease/purchase of a BWR/6 control room simulator (originally intended for the cancelled Black Fox (Okla.) facility) is underway. This simulator will be installed in the TTC building and will be utilized in the BWR courses taught at the TTC. Instruction time on other simulators was leased from TVA and other utilities to provide instruction to the NRC staff on control room operations.

Cooperation with the States

CHAPTER 9

The NRC's contacts with regional, State and local agencies for purposes other than inspection and enforcement or emergency planning are administered through NRC's Office of State Programs. (Certain elements of NRC's State programs are implemented by the Regional Offices under policies and procedures established by the Office of State Programs.) This chapter reports on activities in three major areas: the State Agreements Program; various liaison and cooperative programs; and financial protection and related concerns.

STATE AGREEMENTS PROGRAM

By formal agreement with the Nuclear Regulatory Commission, a total of 27 States have assumed regulatory responsibility over byproduct and source materials and small quantities of special nuclear material. At the end of fiscal year 1985, there were about 13,800 radioactive material licenses in these Agreement States; they represent about 60 percent of all the radioactive materials licenses in the United States. (See Agreement States map on the next page.) The NRC State Agreements Program is implemented by the NRC Regional Offices in accordance with policies and procedures established by the Office of State Programs. In August, 1985 the Governor of Iowa submitted a proposal for an Agreement covering byproduct, source and special nuclear material in less than critical mass quantities. The proposal did not request authority to regulate mill tailings or commercial low-level radioactive waste disposal. The requested effective date for the Agreement is January 1, 1986. The NRC staff prepared an assessment of the proposal which was published in the *Federal Register* for public comment. A decision is expected prior to January 1986.

Review of State Regulatory Programs

The NRC is required by the Atomic Energy Act of 1954 to periodically review Agreement State radiation control programs and confirm that they are adequate to protect public health and safety and are compatible with NRC programs. The reviews follow guidelines contained in a Commission Policy Statement published in the *Federal Register* on December 4, 1981. Any problems identified in these reviews are brought to the attention of State authorities with recommendations for corrective action. Twenty-two routine program reviews, one follow-up review and two special reviews were conducted in 1985. As part of the program review, the NRC technical staff accompanies State inspectors to State-licensed facilities to evaluate inspector performance and to review selected license and compliance casework in detail. One follow-up review of several previously identified program deficiencies was conducted in California in 1985. A special review was conducted in New Mexico and included discussion of the Agreement State program with new staff members in the Environmental Improvement Division; discussions were also held in New York of actions planned by the Department of Environmental Conservation to improve their radiation control program.

The overall conclusion from the NRC reviews conducted during the report period is that the Agreement States continue to conduct effective regulatory programs. Periodic meetings are held with U.S. Department of Labor officials to exchange information and to keep them apprised of the status of Agreement State radiation control programs.

NRC Technical Assistance to States

The NRC provided technical assistance to Agreement States during 1985 in the areas of licensing, inspection, enforcement and proposed statutes and regulations. For example, assistance was provided to Georgia and Utah in their evaluation of license applications for large irradiators utilizing megacuries of radioactivity, to Tennessee in its evaluation of an application for the use of tritium at a proposed Isotope Application Technology facility at Oak Ridge, to Florida in its evaluation of an application for a proposed computerized nuclear cardiology facility, to the New York State Health authorities in their evaluation of americium-241 contamination in a landfill and sewage treatment incinerator resulting from operations of a former State licensee. Assistance was also provided to New York City in its evaluation of a threat (which did not materialize) to contaminate drinking water supplies with plutonium trichloride.

Training Offered by NRC

State radiation control personnel regularly attend NRC-sponsored courses to improve their technical and administrative skills and, thus, their ability to maintain high quality regulatory programs. In 1985, the NRC sponsored 18 short-term training courses, attended by 304 State personnel. Courses included health physics, industrial radiography safety, nuclear medicine procedures, introduction to licensing practices, inspection procedures, well logging, uranium mill inspection, teletherapy calibration, transportation and radiation protection engineering. On-the-job training in licensing and compliance was provided to individual staff members in California, Washington, Arizona and Utah.

Annual Agreement State Meeting

The annual meeting of Agreement State radiation control program directors was held in October 1985 in Bethesda, Maryland and covered a wide range of regulatory issues being faced by State personnel, such as low-level waste, radiation litigation, materials licensing and compliance, revision of regulations, and cases involving radioactive material in unauthorized places.

Irradiation Facilities

Recent regulatory changes related to food irradiation have been proposed and implemented by the Food and Drug Administration. The changes are likely to bring about a significant increase in the irradiation of food by the use of large irradiators. At present, this kind of equipment is used mainly for the sterilization of medical products, using cobalt-60. In view of the expected increase in food irradiation, the NRC sponsored a workshop for NRC and Agreement State staff personnel on regulation of large pool-type irradiators. Department of Energy and Department of Labor representatives also participated. The workshop focused on licensing, construction quality assurance, source loading inspection, pre-operational inspection and initial and routine inspections. The safety considerations involved in using cesium-137 capsules being leased to industry by the Department of Energy were also discussed. A workshop report has been prepared (NUREG/CP-0073) to serve as a radiation safety reference for irradiator designers and operators and for regulatory staffs involved in licensing, inspecting and developing safety standards for such facilities.

Regulation of Low-Level Waste

The NRC continues to provide technical assistance to the Agreement States in their programs for regulating low-level radioactive waste. NRC provided technical assistance to California in its evaluation of a proposed low-level waste disposal site. Assistance was also provided to Washington in the renewal of the U.S. Ecology license for a low-level waste disposal site, to North Carolina in its evaluation of a commercial incinerator and to Nevada in its evaluation of a closure plan for the Beatty commercial low-level disposal site. In addition, South Carolina and Washington are participating in the NRC review of several topical reports on high integrity containers, waste solidification processes and computer codes, used in implementing 10 CFR Part 61.

Regulation of Uranium Milling

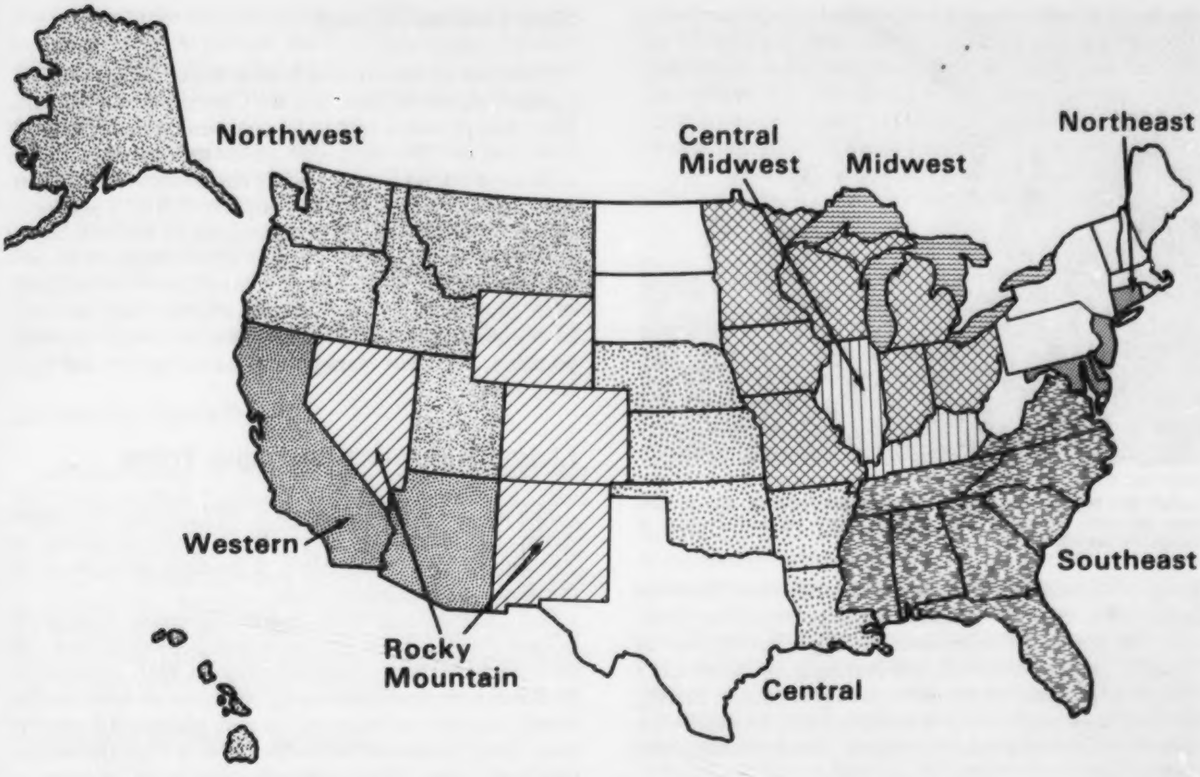
The NRC is continuing to provide assistance to the Agreement States in their programs for regulating uranium milling. This assistance has included guidance on implementing the Environmental Protection Agency's (EPA) ground water requirements; arranging for direct technical assistance on specific cases in the States of Texas, Washington, Colorado, and New Mexico; and arranging for specialized training for the mill regulatory staff for the States of Washington and New Mexico.

COOPERATIVE ACTIVITIES

Low-Level Radioactive Waste Compacts

In accord with the Low-Level Radioactive Waste Policy Act, enacted in December 1980, the States continued their efforts to bring about workable interstate compacts that would provide for regional low-level waste disposal sites. (For map of Compact groups see the *1984 NRC Annual Report*, p. 125.) Seven compacts were proposed to the Congress for the required Congressional consent—Central, Central-Midwest, Midwest, Northeast, Northwest, Rocky Mountain and Southeast.

Students from various states attend NRC training courses to improve their skills in radiation-control activities. These students are taking part in a laboratory exercise during one of the 85 courses conducted during 1985.



LOW-LEVEL WASTE COMPACT GROUPS

Congressional hearings on the matter in 1985 focused mainly on the results of negotiations between those regions with operating disposal facilities and those States and regions without such facilities for continued access beyond the exclusionary date in the Low-Level Radioactive Waste Policy Act: January 1, 1986. These negotiations led to proposed amendments to the Act to assure continued access. The amendments also include provisions intended to resolve outstanding issues affecting all the groups, such as a standard definition of low-level radioactive waste. Issues highlighted by the NRC in its Congressional testimony include the necessity of assuring that there is a home for all radioactive waste, the regulatory uncertainties related to mixed waste disposal (i.e., waste that contains radioactive material regulated by NRC and hazardous waste regulated by the EPA), regulatory problems related to rulemaking for alternatives to shallow land burial technology, and the need for certification by a State that it will manage waste generated in that State after 1992 in the event that an application has not been submitted by the 1990 deadline for the construction of a disposal facility for such waste. Two other issues of primary concern to NRC are the preservation of the NRC's and the Agreement State's regulatory role and the management versus disposal responsibilities exercised by the Compact commissions.

Spent Nuclear Fuel Transportation Seminar

A seminar on the regulation of transportation of spent nuclear fuel was sponsored by the NRC and the U.S. Department of Transportation (DOT) in Chicago, July 31 to August 2, 1985; the 275 participants included State officials, representatives of localities and Indian tribes. The seminar focused on the regulatory roles and responsibilities of NRC and DOT, and on the inter-relation between these Federal activities and State, local and Indian activities with respect to spent fuel transportation. Topics included transportation requirements, routing, shipping and inspection activity, and emergency response. The seminar included a half-day tour of the General Electric Company's spent fuel storage facilities at Morris, Ill. There were also exhibits of the IF-300 rail shipping cask, radiological monitoring vans from NRC Region III and the Illinois Department of Nuclear Safety, the Illinois State police hazardous materials highway patrol car, and escort vehicles from Burns Security and the Northern States Power Company. There were discussion groups on selecting alternate highway routes, effective inspections and emergency preparedness and response. The discussion group on routing identified route selection, risk and accident minimization, and coordination among the States as being the most critical concerns. The inspections discussion

Above is a group of students making a radiation survey of a low-level waste shipment, during an NRC training course on transportation of radioactive materials.

group recommended a definition of the role of each State and Indian tribe, the development of uniform inspection procedures, the use of DOE experience, funding under the Nuclear Waste Policy Act Trust Fund, and sharing of information in a formalized program among all levels of government. Finally, the emergency preparedness discussion group focused on the need for effective plans and procedures, effective training, and equipment.

Memoranda of Understanding

Under a sub-agreement on low-level waste inspection developed by the NRC, States are allowed to inspect waste packaging and shipping procedures on the premises of certain NRC licensees. The inspections cover compliance with State laws and regulations, as well as compliance with NRC's rules and regulations regarding packaging and transportation of low-level waste destined for disposal at a commercial low-level radioactive waste disposal site. The sub-agreement was drawn up in response to State recommendations during formulation of low-level waste compacts.

The first such sub-agreement was signed by NRC and the State of Illinois in June 1984. Negotiations for similar sub-agreements are under way with the States of Pennsylvania, Ohio and Virginia.

The NRC (then AEC), in April of 1967, entered into a Memorandum of Understanding (MOU) with the State of Louisiana which allowed the State to conduct inspections of NRC-licensed materials facilities in offshore waters (within Federal jurisdiction). A similar MOU is presently being negotiated with the State of Texas. All enforcement actions under the MOUs will be carried out by the NRC.

The Office of State Programs continues to seek opportunities to employ the MOU in setting forth principles of cooperation and communication between the NRC and the States.

State Liaison Officers

There are 51 Governor-appointed State Liaison Officers, representing the 50 States and the Commonwealth of Puerto Rico, who provide a contact for communication between the States and the NRC. Both regional and national State Liaison Officers' meetings are periodically held to keep the State Liaison Officers updated on major aspects of NRC's programs. During the fiscal year, regional meetings for the State liaison officers were held in NRC's Region III (Chicago), in December, 1984, and in NRC's Region V (San Francisco), in April, 1985. Subjects discussed at these regional meetings included, among others, emergency response; regional programs; transportation; and waste management, including low- and high-level waste.

Liaison With American Indian Tribes

The Nuclear Waste Policy Act of 1982 provides for Federal Agency consultation and cooperation not only with States, but also with affected Indian Tribes, in reaching decisions on the management of high-level waste.

With the assistance of the National Congress of American Indians (NCAI, which represents over 200 Tribes) and with the Council of Energy Resource Tribes (CERT, representing 39 Tribes with significant energy resources on their reservations), a number of Indian issues were addressed during the year. Transportation of radioactive waste is of particular concern to the tribes. Those tribes identified by NCAI as having had radioactive waste transported through their reservations were invited to attend a DOT/NRC Spent Nuclear Fuel Transportation Seminar held in Chicago, July 31- August 2, 1985. (See above.) Sixteen representatives of 12 Tribes and NCAI participated in the transportation seminar.

INDEMNITY, FINANCIAL PROTECTION AND PROPERTY INSURANCE

The Price-Anderson System

Under NRC regulations implementing the Price-Anderson Act, a three-layered system was set up to pay public liability claims in the event of a nuclear incident causing personal injury or property damage.

The first layer requires all licensees of commercial nuclear power plants rated at 100 electrical megawatts or more to provide proof of financial protection in an amount equal to the maximum liability insurance available from private sources. Currently, this amount is $160 million.

The second layer provides for a retrospective premium payment mechanism whereby the utility industry would share liability for any damages resulting from a nuclear incident in excess of $160 million. In the event of such an incident, each licensee of a commercial reactor rated at 100 electrical megawatts or more would be assessed a prorated share of damages up to the statutory maximum of $5 million per reactor per incident. At present, the secondary financial protection layer is $480 million (derived from 96 power reactors rated in excess of 100 MW(e) licensed to operate times $5 million per reactor).

The third layer—Government indemnity—had formerly amounted to the difference between the $560 million limit of liability and the sum of the first and second layers. Government indemnity for reactors was phased out for large power reactors, however, on November 15, 1982, when the sum of the first and second layers reached $560 million. The limit of liability for a single nuclear incident now increases without limit in increments of $5 million for each new commercial reactor licensed.

Indemnity Operations

As of September 30, 1985, 141 indemnity agreements with NRC were in effect. Indemnity fees collected by the NRC from October 1, 1984, through September 30, 1985, totalled $19,136. Fees collected since the inception of the program total $23,117,902. Future collections of indemnity fees will continue to be lower since the indemnity program has been phased out for commercial reactor licensees. No payments have been made under the NRC's indemnity agreements with licensees during the 28 years of the program's existence.

Insurance Premium Refunds

The two private nuclear energy liability insurance pools—American Nuclear Insurers and the Mutual Atomic Energy Liability Underwriters—paid policyholders the 19th annual refund of premium reserves under their Industry Credit Rating Plan. Under the plan, a portion of the annual premiums is set aside as a reserve either for payment of losses or for eventual refund to policyholders. The amount of the reserve available for refund is determined on the basis of loss experience of all policyholders over the preceding 10-year period.

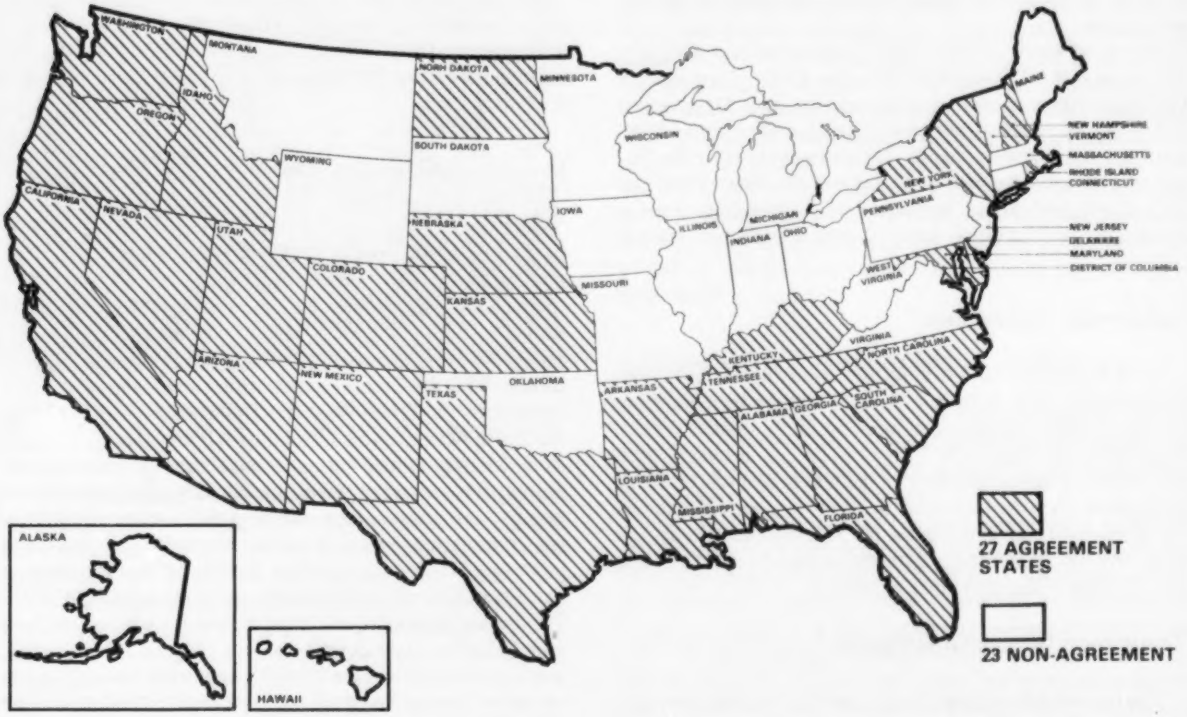
Refunds paid in 1985 totalled $4,945,796—approximately 47 percent of all premiums paid on the nuclear liability insurance policies issued in 1975 and covering the period 1975-1985. The refunds represent 74.1 percent of the premiums placed in reserve in 1975.

Price-Anderson Renewal

In December 1983, the Commission transmitted to the Congress a detailed report, NUREG-0957, entitled *The Price-Anderson Act-The Third Decade*, concerning the need for renewal or modification of the Price-Anderson Act, which will expire on August 1, 1987. (For background, see the *1983* and *1984 NRC Annual Report*, p. 104 and p. 127 respectively). On June 4, 1985, the Commission testified before the House Subcommittee on Energy and the Environment on the Price-Anderson Act and on the Commission Price-Anderson Report. At the hearing—where testimony was also heard from representatives of the insurance and utility industries and public interest groups—the primary focus was on a decision by a majority of the Commission to support an industry-wide limitation on liability of $2-to-$5 billion, rather than the annual limitation on liability of $10 millin per reactor, proposed in NUREG-0957. The annual limitation continued to have the support of Chairman Palladino and Commissioner Asselstine. The Commission also testified on June 25, 1985, before the Senate Subcommittee on Energy Research and Development and on October 23, 1985, before the Senate Subcommittee on Nuclear Regulation. The subject of those hearings was S.1225, a bill introduced by Senators Simpson and McClure which would make a number of major changes in the Price-Anderson Act while still retaining an absolute liability limit.

Representatives from States, Indian Tribes (12 different tribes participated) and localities were briefed on the characteristics of a spent fuel cask at the General Electrical Company's Morris, Ill., facility during a joint NRC/DOT-sponsored seminar on transportation of spent fuel. The seminar was held in Chicago in early August of 1985.

NRC STATE AGREEMENTS PROGRAM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TENNESSEE
ARKANSAS
TEXAS
SOUTH CAROLINA
ALABAMA
GEORGIA
LOUISIANA
MISSISSIPPI
FLORIDA
ALASKA
HAWAII
▨ 27 AGREEMENT STATES
☐ 23 NON-AGREEMENT

Financial Qualifications Reviews of Electric Utilities

NRC rules (10 CFR 50.33(f)) and Appendix C to 10 CFR Part 50) provide for pre-licensing financial qualifications reviews of electric utilities that apply for power reactor construction permits. (For background, see the *1984 NRC Annual Report*, p. 127.) As discussed in the last year's annual report, the NRC had amended its rules to eliminate pre-licensing financial qualifications reviews and findings regarding electric utilities applying for power reactor operating licenses. At the close of the report period, the present financial qualifications rule—which provides for financial reviews of electric utilities at the construction permit stage, but not at the operating license stage—was under challenge in a case pending before the U.S. Court of Appeals for the D.C. Circuit. (*New England Coalition on Nuclear Pollution, et al. v. NRC*, D.C. Circuit Case No. 84-1514.) The petitioners seek to have the rule remanded to NRC for a further rulemaking proceeding. Oral argument was heard before the appeals court on October 11, 1985.

Incentive Regulation of Electric Utilities

Performance incentives established by State public utility commissions (PUCs) are applicable to the construction or operation of about 35 reactors licensed for 25 investor-owned utilities in 15 States. Incentive mechanisms are used to measure a utility's efficiency level in operation or construction of generating plants and to financially reward or penalize the utility for performance above or below established levels. The purpose of performance incentives, of course, is to encourage sustained improved performance, but there is concern over the possible effects on safety of such incentives. The concern is that, in the interest of short-term economics, pressures may cause utilities to take short cuts, delay shutting down a reactor or take some similar action in order to meet a deadline or to avoid a cost limitation or other penalty. Because of this concern, the NRC staff has begun monitoring performance incentives applicable to nuclear plants. The staff has also begun developing a plan to analyze possible safety concerns at specific plants and to disseminate information to its Regional Offices.

Property Insurance

On November 8, 1984, the Commission published a proposed rule in the Federal Register (49 *FR* 44645) which would, if promulgated, increase the amount of on-site property damage insurance from the current minimum of $585 million to $1.02 billion. The NRC believes that such insurance should be required so that the financing and the pace of clean-up following an accident does not become a public health and safety

As work continued on the cleanup of Three Mile Island Unit 2, the parent company, General Public Utilities Corp., announced that some $300 million of insurance proceeds, plus $6 million in interest thereon, had been spent on the effort, as of the end of 1984.

problem. The main issues addressed in the proposed rule were (1) whether the Federal government can preempt State law that prohibits certain public utilities from buying insurance offered by mutual companies or insurance requiring payment of a retrospective premium, and (2) whether a priority in the payment of insurance proceeds for decontamination and clean-up should be imposed.

The NRC received 35 comments on the proposed rule. Most comments were directed to the issues of state prohibitions against the purchase of certain types of insurance and whether a decontamination priority should be imposed. Relatively few comments expressed concern with the amount of insurance proposed to be required. The staff has prepared a final rule which addresses the comments submitted. This final rule will be considered by the Commission during the fall of 1985.

The third annual property insurance reports submitted by power reactor licensees indicate that of 66 sites insured as of April 1, 1985, 42 carried $1.085 billion in coverage and another 10 carried at least $1.0 billion. Four other sites have been exempted from the requirement to carry insurance in excess of $500 million because of size or design considerations. As of July 15, 1985, Nuclear Electric Insurance Limited (NEIL-II) announced that the amount of excess property insurance available from them would be increased to $525 million, bringing the total generally available from all insurers to $1.11 billion.

STATUS OF TMI-2 FACILITY

Financial Aspects of TMI-2 Cleanup

Funding by GPU. (For background, see the *1984 NRC Annual Report* pp. 127-128.) Property insurance proceeds totalling $300 million plus $6 million of interest thereon had been expended on cleanup of the damaged facility at Three Mile Island Unit 2 (TMI-2, Pa.) by the end of 1984. These expenditures fully exhausted available insurance proceeds.

Revenues collected by General Public Utilities Corporation's (GPU) three operating subsidiaries in Pennsylvania and New Jersey continued to be expended on cleanup during 1985, at the total annual rate of $34 million. Existing rate orders provide that customer funding of cleanup will increase to $49 million on an annual basis following restart (and coincident with commercial operation) of TMI-1. Restart was accomplished in October 1985. GPU continues to provide cash advances from internal sources to alleviate any cash flow problem related to cleanup. The total 1985 advance is estimated at $11 million. The GPU projections provided to NRC indicate a continuing GPU commitment to provide such cash advances as needed for the cleanup. Continued improvement in GPU's financial condition and cash flow position gives greater assurance that such cash advances will be made.

Cost Sharing Plan. During 1985, GPU announced that it had received not only commitments but also cash payments from all suggested contributors in the TMI-2 cleanup cost sharing plan proposed by Pennsylvania Governor Richard Thornburgh in July 1981. By November 1985, the Edison Electric Institute's (EEI) industry cost-sharing program had paid its committed $25 million annual contribution for 1985, the first year of industry contributions through the EEI program. Payments by EEI member investor-owned utilities are being supplemented by grants from Pennsylvania and New Jersey utilities so that a total of $150 million will be provided for cleanup by the EEI program over a period of six years.

Contributions under the cost sharing plan continued to flow in 1985—from the State governments of Pennsylvania and New Jersey, from the Federal Government through the Department of Energy, from a Japanese industry consortium, and from GPU customers and GPU internal sources as discussed above. During 1985 the Commission found that there appears to be reasonable assurance that there will be adequate funding for the cleanup for the next several years. The Commission indicated that it would continue to monitor closely the cleanup funding situation.

International Programs

CHAPTER 10

The NRC coordinates its international activities through the Office of International Programs. Many of the other NRC offices participate in these activities, contributing technical experts and expertise and conducting research both here and overseas. The purposes of the NRC's international activities are to improve world-wide nuclear safety cooperation and to assist the Government's efforts to ensure against further nuclear-explosive proliferation, especially against any potential contribution from U.S. nuclear exports.

Highlights of Fiscal Year 1985

During the report period, the NRC:

- Renewed bilateral nuclear safety cooperation arrangements with Sweden, the Philippines and Finland, and concluded a new bilateral arrangement with Yugoslavia.
- Continued to expand its network of mutually beneficial agreements on nuclear safety research. To date, the net effect of this agreement program has been that the United States has access to approximately $300 million worth of research for which the U.S. would otherwise have had to pay.
- Dispatched an 11-member team of senior technical staff and ACRS representatives (see Chapter 2) to hold extensive discussions with Japanese safety authorities and to tour Japanese nuclear facilities.
- Arranged for over 130 visits by representatives from governments, and from public and private organizations overseas.
- Participated in the 1985 Non-Proliferation Treaty Review Conference and the International Atomic Energy Agency's (IAEA) annual General Conference.
- Issued 193 export licenses and 41 amendments to existing licenses.
- Participated with the Executive Branch and the Congress in reviewing the proposed U.S.-China Agreement for Peaceful Nuclear Cooperation, negotiated by the State Department.
- Continued to support domestic and international efforts to ensure that the risk of nuclear proliferation is minimized in the activities that make up the nuclear fuel cycle.
- Worked closely with the Executive Branch to assist the IAEA in strengthening international safeguards.

INTERNATIONAL COOPERATION

Bilateral Information Exchange Arrangements

The NRC participates in a wide-ranging and continually expanding program of information exchange and cooperative safety and research activities with its international counterparts. Since May 1974, the NRC has conducted most of its general technical information exchange under the umbrella of a series of nuclear safety arrangements made with the regulatory authorities of the following countries and areas of the world: Belgium, Brazil, China, Denmark, Egypt, Finland, France, the Federal Republic of Germany, Greece, Israel, Italy, Japan, Korea, Mexico, The Netherlands, the Philippines, Spain, Sweden, Switzerland, the United Kingdom, Yugoslavia, and Taiwan. These 22 arrangements are effective for a period of five years from their date of signature, but contain provisions for renewal by mutual agreement. They establish a mechanism for the timely exchange of significant reactor safety information and set up official communications channels for the sharing of information on broad reactor safety problems and other matters of mutual interest. They also serve as the foundation for most of the nuclear safety assistance that the NRC is able to provide to developing countries, particularly to those using U.S. reactors and other U.S. equipment.

The arrangements with Sweden, the Philippines, and Finland were renewed during this reporting period, and a new arrangement with the Yugoslav Federal Committee for Energy and Industry was completed. Negotiations on the renewals of existing arrangements with Japan and Denmark are under way.

Bilateral and Multilateral Safety Research Agreements

The NRC has also established a network of general and specific agreements on research cooperation which has grown in both depth and substance over the last few years. The NRC currently has over 50 agreements with 14 countries under which it cooperates in ongoing nuclear safety research projects both in the U.S. and overseas. These research projects make direct contributions of data and analyses needed to confirm and assess computer codes used in the NRC licensing and regulatory process.

Some of the cooperative research involves experiments performed in U.S. facilities, with foreign entities making direct cash contributions (as high as $5 million apiece) to participate.

In September 1985 in Vienna, the NRC and its Finnish counterpart organization renewed a long-standing agreement on the exchange of nuclear safety information. Shown here signing the renewal arrangement are (seated) NRC Chairman Nunzio J. Palladino (left) and Chief Director Antti P. U. Vuorinen of the Finnish Center for Radiation and Nuclear Safety. Standing at left is J. R. Shea, Director of the NRC Office of International Programs and at right is Finland's Undersecretary of State for Political Affairs, Klaus Tormudd.

Some of it entails the foreign country's providing the results of other complementary or supplementary research it has sponsored in an area of interest or need to the U.S. In other cases, experiments are performed overseas, and the U.S. either contributes instrumentation, test analysis and on-site experts, or makes direct cash contributions to the host program. Some of the programs are purely analytical, with from two to 10 countries contributing towards the assessment of safety analysis computer codes, using their experts, computers and data from their own test facilities. This has the benefit of assuring that the computer codes give realistic, consistent answers to regulatory questions on a range of nuclear accidents as analyzed by different people in varying environments.

Regulatory Exchange Visit to Japan

Japan requires that its licensees conduct extensive inspection and maintenance on each of their facilities on an annual basis. This requirement is credited as one of the principal factors behind Japan's impressive record for overall plant availability, which has risen from nearly 55 percent in 1974 to nearly 72 percent in 1983, and over 74 percent in 1984, with a very low average reactor scram frequency (0.1 in 1984). In order to study this impressive program, the NRC dispatched an 11-member team of senior technical staff and ACRS representatives in late 1984 to hold extensive discussions with Japanese safety authorities and to tour Japanese facilities. Information and safety experience were exchanged in a number of important areas—including plant maintenance, inspection techniques, human factor-related issues, diesel generator reliability, and operating experience. It is anticipated that the information and lessons learned from the Japanese program will have an important influence on U.S. regulatory activities in the years to come. Since this visit, NRC management has strongly encouraged U.S. utilities to develop exchange programs with their counterparts in Japan.

ACRS Visit to France and Germany

Six members of the Advisory Committee on Reactor Safeguards (ACRS) visited their counterparts in France, the Group Permanent Reactor (GPR), and in Germany, the Reaktor-Sicherheitskommission (RSK), for discussions of light water reactor technical issues and site visits.

In France, discussion topics included the safety, availability and reliability of a standardized plant design; improved pilot-operated valves; control room design; and station blackout. The team also visited the Paluel facility where the first of the 1300 MWe reactors of advanced French design had recently started operation.

In the meeting with the German RSK, both sides discussed their operating experience with LWRs, the impact of operating experience on maintenance, and early failure detection that includes computer-assisted acoustical testing for vibration monitoring, loose particle monitoring and leakage detection.

International Emergency Preparedness Cooperation

During the year, practical arrangements were completed for the NRC to give technical advice to the Taiwan regulatory authority in the event of a radiological emergency at one of Taiwan's three nuclear sites. NRC's role would be to offer regulatory advice, if requested by its counterpart agency, on questions concerning U.S. equipment or U.S.-derived procedures at the foreign plant. A similar arrangement for cooperation during radiological emergency situations was made with the Korean Ministry of Science and Technology in November 1981.

The NRC is in the final phases of preparing program guidance for international cooperation during radiological emergencies. The completed document will outline the scope, application and limitations inherent to cooperation in this area.

Technical Safety Cooperation

During the report period, over 130 requests were received from foreign governments and public and private organizations concerning visits to the NRC. The Office of International Programs coordinates visit requests with appropriate staff offices and individuals to address discussion topics and to promote a meaningful two-way flow of information. In addition to these visits, the NRC provided responses to nearly 100 requests for technical and safety information. Documents and publications are also routinely distributed to our foreign partners in regular mailings.

Foreign Assignees to the NRC Staff

Strong interest continued in the NRC's on-the-job training program for foreign nationals. This year the NRC hosted 21 foreign assignees from 10 different countries. Their primary areas of interest included risk analysis, radiological protection, electrical instrumentation and control, human factors, reactor and containment systems, emergency planning and response, and legal aspects of regulation. The assignees served in the NRC's Offices of Nuclear Reactor Regulation, Nuclear Regulatory Research, Inspection and Enforcement, and several of the NRC Regional Offices. Many of them also took the opportunity during their long-term assignments to attend technical training courses at the NRC's Training Center.

PARTICIPATION IN INTERNATIONAL ORGANIZATIONS AND CONFERENCES

Activities in the OECD

The NRC participates in the 24-nation Organization for Economic Cooperation and Development (OECD) through its membership in the Nuclear Energy Agency (NEA), one of the OECD's specialized agencies. This agency brings together specialists from Western Europe, the United Kingdom, Canada, Japan, and the U.S. to exchange information and coordinate joint activities on nuclear technology and safety-related issues. Two of the major committees of the NEA were chaired by NRC representatives in 1985: William J. Dircks, Executive Director for Operations, served his third year as Chairman of the Committee on the Safety of Nuclear Installations (CSNI) and Richard E. Cunningham served his second year as Chairman of the Committee on Radiation Protection and Public Health.

Safety Assistance in the IAEA

During 1985 the NRC, in coordination with the International Atomic Energy Agency in Vienna, Austria, continued its pro-

Operational safety and research information and technical assistance have been exchanged between the United States and Japan since the latter first entered the nuclear reactor field. In late 1984, an NRC/ACRS team toured Japanese facilities to learn more about that country's impressive records in such areas as plant availability and reactor scram (shutdown) frequency. In October 1985, a Japanese team visited NRC headquarters to discuss research cooperation. Among those pictured above during the latter visit are Tsuneo Fujinami, President of the Japan Atomic Energy Research Institute (fourth from the left), NRC Chairman Nunzio J. Palladino (at center) and Commissioner Lando W. Zech (to the right of the Chairman).

gram of providing nuclear safety advice and assistance to member countries initiating or developing their nuclear power programs. NRC sent two advisers to Korea, one, for two months, to advise on waste disposal and the other, for one week, to advise on nuclear reactor material surveillance techniques. In September 1985, an NRC inspector completed a one-year assignment in the Philippines, during which he advised the Philippine Atomic Energy Commission on how to develop its regulatory inspection program. The NRC also provided advisers to Mexico in the areas of final safety analysis report evaluation and construction and quality assurance as well as two instructors to conduct a two-week course on BWR Fundamentals. A two-week PWR Fundamentals course was taught in China by two instructors, and NRC staff members participated in two short-term missions there on ASME code requirements and severe accident research. An NRC adviser was sent to Yugoslavia on two separate safety missions to advise on human factors and abnormal and emergency procedures. NRC personnel conducted a number of short-term nuclear safety workshops in Egypt this year and gave staff support to two IAEA/Argonne National Laboratory-sponsored training courses in Korea and China.

IAEA General Conference

NRC was represented at the 29th IAEA General Conference (September 23-27, 1985) in Vienna, Austria, by Chairman Nunzio J. Palladino, Executive Director for Operations William J. Dircks, and Director of International Programs James R. Shea. Chairman Palladino and Mr. Dircks also participated in a nuclear safety technical session with other senior regulators in attendance at the Conference. This was the second such session held at the General Conference.

1985 Non-Proliferation Treaty (NPT) Review Conference

The third NPT Review Conference was held from August 27 to September 21, 1985, in Geneva, Switzerland, with the NRC participating as a member of the U.S. Delegation. The conference concluded with all countries agreeing on a final document and all parties continuing their support for the objectives of the Treaty. The Conference noted the IAEA's considerable level of effort in the area of technical assistance and urged that such cooperation be increased wherever possible.

EXPORT-IMPORT ACTIONS

NRC Export License Summary for Fiscal Year 1985

During the fiscal year ending September 30, 1985, the NRC issued 193 export licenses and 41 amendments to existing licenses. Of the licenses issued, 88 were "major" licenses in four categories: special nuclear material, source material, components, and nuclear reactor materials. Most major licenses involved routine exports of low-enriched uranium intended for use in commercial light-water power reactors. Eight licenses involved exports of high-enriched uranium to research reactors (271 kilograms) and the German Thorium High Temperature Reactor (52 kilograms). A total of 15 nations received shipments of special nuclear material under major export licenses during the year. As in the previous year, export licenses were issued to the European Atomic Energy Community (EURATOM) for major quantities of source material for enrichment and subsequent power reactor use. The remaining 105 export licenses included 21 for small quantities of special nuclear materials, 11 for source material, 18 for byproduct material, and 55 for components and materials.

Revisions to NRC's Export Licensing Regulations

In January 1985, the NRC implemented several revisions to the Commission's export licensing regulations (10 CFR Part 110). The amended regulations expand the authority of exporters to export non-sensitive nuclear equipment and minor

Five international assignees to the Division of Systems Integration and Human Factors in the NRC Office of Nuclear Reactor Regulation (NRR) discuss their countries' nuclear power programs with NRR Director Harold R. Denton, second from left. W. LaVine, at left, of the NRC International Programs staff, assists assignees in arranging their work and study programs at the NRC. The assignees are, beginning to the right of Mr. Denton, J. L. Milhem from France, A. Pawlak from Poland, C. Balatbat from the Philippines, P. Koutaniemi from Finland and J. Basurto from Mexico.

Dr. Samuel A. Harbison of the United Kingdom Nuclear Installations Inspectorate is shown briefing the NRC staff on the British approach to safety goals for nuclear facilities. The briefing and discussion took place during Dr. Harbison's visit to NRC staff offices in Bethesda, Md., in the fall of 1985.

quantities of nuclear material without applying for or obtaining a specific NRC export license. These amendments have reduced significantly the total number of required licensing actions without affecting the Commission's existing rigorous controls over the export of proliferation-sensitive nuclear commodities.

Subsequent Arrangements and Other Export Consultations With the Executive Branch

In addition to its own licensing actions, the NRC consults with the Executive Branch on other types of transactions with potential proliferation implications. These transactions include nuclear technology transfers and certain subsequent arrangements and agreements for cooperation. A significant number of these transactions involve export cases licensed by the Department of Commerce. The NRC was consulted on over 300 of these cases during fiscal year 1985.

Also during this period, the NRC reviewed 72 Executive Branch requests for subsequent arrangements. These arrangements describe further actions that a country wishes to take with previously exported U.S.-origin nuclear material, equipment, facilities, or technology. Included in these requests were several cases involving now routine retransfers of spent U.S.-origin nuclear fuel from Japan and Switzerland for reprocessing in the United Kingdom and France. Three consultation cases required considerable Commission involvement. The first concerned the employment of several U.S. citizens on projects related to the operation of the Koeberg Reactors in South Africa. These U.S. citizens had not received prior U.S. authorization (from the Department of Energy) to work in nuclear related projects in South Africa (as required by 10 CFR 810). After learning of this requirement, the U.S. citizens requested appropriate authorization. Their requests were reviewed by several government agencies, including the NRC. These requests provoked a good deal of Congressional interest, resulting in further requests for NRC action on this matter. Most of the requests were denied by the Department of Energy.

The second matter was the review of the proposed U.S.-China Agreement for Peaceful Nuclear Cooperation. The Agreement was first reviewed in 1984, and the NRC provided its final views to the President in July 1985. While the NRC did not object to the Agreement, the Commission expressed some concern with certain aspects of it and was asked to testify at several Congressional hearings on their concerns. At the close of the report period, the Agreement was still undergoing Congressional review.

Finally, the NRC and the Executive Branch reviewed a request to extend the joint U.S.-Japan determination that reprocessing of U.S.-origin material can be effectively safeguarded at Japan's Tokai-Mura reprocessing facility. This determination was originally agreed to in 1977 and subsequently renewed several times. A significant aspect of this request was the Executive Branch analysis of "timely warning." Timely warning is a major element of the analysis required by U.S. law for each subsequent arrangement request; it concerns the question of whether the U.S. Government will be able to respond in time to prevent a country from using U.S.-controlled nuclear material to build and explode a nuclear explosive. The NRC has been involved in discussions with the Executive Branch for some time concerning the adequacy of its analysis of timely warning. The Japanese Tokai-Mura case has provided another opportunity for the NRC and the Executive Branch to attempt to resolve their differences over this issue.

International Safeguards and Physical Security

For all pending export cases licensed by the NRC, the staff reviews the implementation of the International Atomic Energy Agency (IAEA) safeguards and physical security arrangements to be applied to the exports in the receiving country. These reviews are performed in compliance with U.S. nonproliferation laws to ensure that U.S. exports will be protected during transit and use in the importing country and will not be used for proscribed purposes, such as the making of nuclear explosives.

With respect to international safeguards, the NRC participates in U.S. Government efforts to assist the IAEA in improving its safeguards system. The primary programs in this area are the U.S. Program of Technical Assistance to IAEA Safeguards (POTAS) and the U.S. Action Plan Working Group (APWG). Through the activities of these groups, the U.S. is also able to take part in joint projects with other countries in support of the IAEA. In 1985, the NRC, along with other Federal agencies, participated in bilateral and multilateral safeguards discussions and research projects with Japan, France, the United Kingdom, West Germany, and the European Community.

The NRC has substantial responsibility for implementing the U.S./IAEA Agreement to apply international safeguards to selected U.S. nuclear facilities. The NRC participates in negotiation of the arrangements for applying international safeguards on facilities it licenses. In addition, the NRC assists the IAEA in scheduling and organizing its inspection activities at NRC-licensed plants and accompanies the inspector during the inspection. In 1985, three NRC-licensed facilities were subject to international safeguards—Arkansas 2, San Onofre 2 (Cal.), and the Combustion Engineering Low Enriched Uranium Fuel Fabrication Plant.

Nuclear Regulatory Research

CHAPTER 11

The NRC's Office of Nuclear Regulatory Research (RES) provides research information essential to these vital regulatory tasks: to help create an adequate technical basis for the rulemaking and regulatory decisions which support NRC licensing and inspection activities, to assess the feasibility and effectiveness of safety improvements, and to increase basic understanding of those phenomena for which analytical methods are indispensable in framing effective regulation.

The Office also has the responsibility for developing and coordinating NRC standards—the regulations and guides governing licensed activities of the United States nuclear industry. Regulations are set forth in Title 10, Chapter I, of the Code of Federal Regulations and are published in the *Federal Register*. Those produced by the NRC in 1985 are listed in Appendix 4. Regulatory guides are described in Appendix 5, which provides a listing of those guides issued, revised, or withdrawn during fiscal year 1985.

OPERATING REACTOR INSPECTION, MAINTENANCE AND REPAIR

Reactor Pressure Vessels

Pressurized Thermal Shock and Vessel Aging Studies. During the last four years, significant research effort has been directed toward resolving the pressurized thermal shock (PTS) safety issue. Under certain potential accident conditions—such as small-break loss-of-coolant accidents, main steam line breaks, steam generator overfilling scenarios, and associated instrument and component failures—a pressurized water reactor (PWR) pressure vessel could be subjected to severe differential cooling rates, coupled with the maintenance of high pressure within the vessel. This combination of thermal stresses and internal pressure stresses, called PTS, could pose a serious challenge to some of the older reactor pressure vessels which, because of prolonged neutron irradiation, have developed a degree of embrittlement.

All prior NRC-sponsored research effort in the area of pressure vessel materials and fracture mechanics—particularly development of experimentally validated analysis methods and irradiation effects studies carried out at the Oak Ridge National Laboratory (ORNL), the Naval Research Laboratory (NRL), and Materials Engineering Associates, Inc. (MEA)—developed data that not only led to the early recognition of the problem but also to its rapid and effective resolution. This resolution took the form of an embrittlement level screening criterion to be applied to reactor pressure vessels. This screening criterion, called the Reference Temperature-PTS, represents an embrittlement level beyond which utilities cannot operate without the permission of the NRC. It was incorporated into the Federal regulations during 1985. An amendment to 10 CFR Part 50 established the screening criterion and also stated that the NRC would issue a regulatory guide for the utilities to follow as their reactor vessels approach the screening limit.

The continuing research effort in this area is twofold: (1) the establishment through confirmatory research of the levels of conservatism built into the screening criterion, and (2) the development of support data on the degradation of reactor pressure vessel toughness and on the fracture mechanics methodology, both deterministic and probabilistic, to be used in the preparation of the regulatory guide on PTS. In keeping with these research directions, a series of large-scale, model pressure vessel experiments was initiated in 1984, with the first PTS experiment at ORNL as part of the Heavy-Section Steel Technology (HSST) program. This was the first controlled experiment, in which a pressure vessel with wall-thickness approaching that of a full-scale reactor vessel was flawed and subjected to combined thermal and pressure transients similar to those that could be encountered during an actual PTS event. This experiment confirmed the theoretically predicted fracture behavior and demonstrated the beneficial effect of the warm prestressing phenomenon. This latter point was particularly important in that warm prestressing is conservatively omitted in the NRC's criteria for evaluating PWRs. The first test vessel was fabricated of present day high-toughness material that was heat treated to make its toughness equivalent to that of moderately embrittled steel.

During 1985, planning was completed and procurement initiated for the second experiment in the series. This experiment, to be conducted during April and May 1986, will be similar to the first experiment except that the vessel material will be of low upper-shelf-Charpy V-notch energy toughness, which is representative of several reactor vessels presently in service. This experiment is specifically designed to validate the applicability of the NRC's PTS criterion to all classes of reactor pressure vessel materials now in service.

Studies supplementary to the PTS vessel experiments are under way at ORNL, the National Bureau of Standards (NBS), and MEA, with significant subcontracts at Southwest Research Institute and the University of Maryland. This work involves wide-plate crack arrest studies, dynamic effects in crack propagation and arrest, and the inhibiting effect of the stainless steel cladding that lines all commercial reactor pressure vessels on

REGULATIONS AND GUIDES

NRC standards are primarily of two types:

- Regulations, setting forth requirements that must be met by NRC licensees in Title 10, Chapter I, of the *Code of Federal Regulations*.
- Regulatory guides, describing, primarily, methods acceptable to the NRC staff for implementing specific portions of NRC regulations.

When NRC proposes new or amended regulations, they are normally published in the *Federal Register* to allow interested persons time for comment before they are adopted. This is required by the Administrative Procedure Act. Following the public comment period, the regulations are revised, as appropriate, to reflect the comments received. Once adopted by the NRC, they are published in the *Federal Register* in final form with the date they become effective. After that publication, rules are codified and included annually in the *Code of Federal Regulations*. Some regulatory guides describe techniques used by the staff to evaluate specific situations. Others provide guidance to applicants concerning the information needed by the staff in its review of applications for permits and licenses. Many NRC guides refer to or endorse national standards (also called "consensus standards" or voluntary standards) that are developed by recognized national organizations, often with NRC participation. The NRC makes use of a national standard in the regulatory process only after an independent review by the NRC staff and after review of public comment on NRC's planned use of the standard.

The NRC encourages comments and suggestions for improvements in regulatory guides and, before staff review is completed, issues them for comment to many individuals and organizations along with the value/impact statements that indicate the objectives of each guide and its expected effectiveness and impact.

the initiation of cracking in those vessels during an overcooling scenario. Of particular interest is the cooperative effort between the NRC and NBS for the wide-plate tests. These tests are intended to develop data to extend the application of fracture mechanics to wider ranges of materials and loading conditions. They include the analysis of large-plate specimens that are deliberately flawed, subjected to severe thermal gradients, and tested to produce long crack propagations and arrest. These crack tests require the use of the largest available loading machine in the country, which is at NBS (see figure). The first test was conducted in September 1984, and four subsequent tests have been carried out in 1985. Data for these experiments, including extremely high strain rate data, represent the state of the art in fracture experimentation and are being used to develop improved analytical models for a better understanding of the margins that current design criteria provide against fracture.

All this research is not only confirming and validating the present methodologies and practices, in order to ensure continued safe operation of present reactors, but it is also laying down the regulatory basis for eventual plant-life extension beyond the original license period.

Radiation Embrittlement and Dosimetry. Normal operation of reactors produces excess neutrons from fuel fissioning which impinge upon the reactor pressure vessel wall, and, dependent upon the constituents in the steel of these walls, cause it gradually to become brittle, to varying degrees over the operational lifetime of the vessel. This problem has been studied for many years at many laboratories here and around the world. Past research has identified certain chemical constituents, whether alloying elements or residual elements of pressure vessel steel, that promote this embrittlement process. Standards for steels to be used for reactor pressure vessel walls that effectively minimize this embrittlement process have been set. Newer reactor vessels now in service were fabricated to these standards, and improvements in steel performance are being noted. The HSST Fourth Irradiation Series was completed in 1985, and the study demonstrates that welds and plate material fabricated to the new standards show a marked resistance to neutron embrittlement, as compared to vessel materials from previous fabrication processes that did not control such chemical elements as copper, nickel, and phosphorus to the same degree. Sufficient specimens were included in the Fourth Irradiation Series to allow a statistical analysis of the results, a procedure lacking in prior work. The HSST Fifth Irradiation Series was initiated in 1984, with irradiation continuing through 1985 and into 1986. This series is designed to validate the Code-designed trend behavior for the irradiation-induced changes in fracture properties that are used to evaluate vessel safety for continued plant operation. In this series, specimens up to 8.0 inches thick (unirradiated) and 4.0 inches thick (irradiated) are being tested to obtain statistically acceptable data in a load and temperature range not previously achieved. All irradiations and testing will be completed by the middle of 1987.

The HSST Sixth Irradiation Series was begun in 1985. This series, using the same material as used in the Fifth Irradiation Series, will allow examination of the effects of irradiation on the crack arrest properties of vessel welds. This work complements the Fifth Series, which deals with crack initiation.

Both at ORNL and at MEA, irradiation effects studies are continuing on the stainless steel cladding material used in present day reactor vessels. This study has direct applicability to the ability of the cladding to inhibit crack initiation and growth during PTS or other accident conditions. Preliminary indications are that "good-practice" stainless steel cladding is highly resistant to irradiation effects (at least for the expected irradiation range for lightwater reactor vessels), while "poor-practice" cladding suffers significant irradiation damage. Efforts to incorporate these data into the already developed fracture mechanics models continue.

In 1985, a research program to develop a mechanistic model for the irradiation damage of reactor vessel steels was begun. This study should allow the NRC to place less reliance upon empirically developed data bases and will contribute to the development of a regulatory basis for establishing plant-life extensions.

Steam Generators

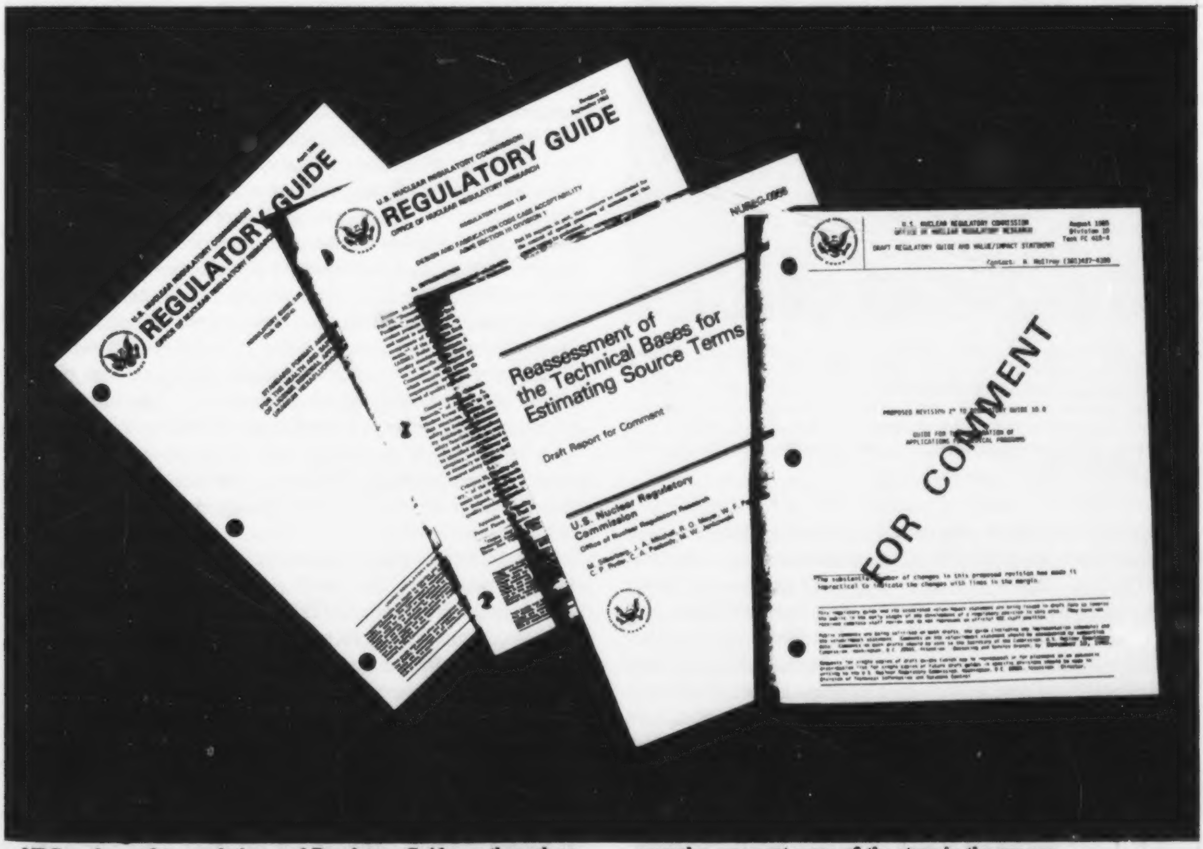
The Steam Generator Group Project at Battelle-Pacific Northwest Laboratories (PNL) works with a steam generator retired from service in an actual PWR facility as its test bed for research on a number of licensing, safety, and reliability issues (see *1984 NRC Annual Report*, pp. 136-137).

In 1985, repair procedures for degraded tubing and anti-vibration bar removal and replacement procedures were demonstrated. During 1984 and 1985, the program concentrated on a series of nondestructive examinations (NDE) of generator tubes and data analyses to determine both the reliability and repeatability of examinations for detecting and sizing various kinds of flaws and also to ascertain the best current methods available for inservice inspections. A large variability in the detection and sizing of various flaws has been noted between inspection teams and analysts. The validation of these methods will be completed in 1986 through removal and destructive metallographic examination and leak rate testing and the bursting of degraded tubes. Based on the correlations obtained from the NDE signals and the destructive examinations, improved tube-plugging criteria and tube inspection plans will be proposed in 1986 for meeting Code and regulatory needs.

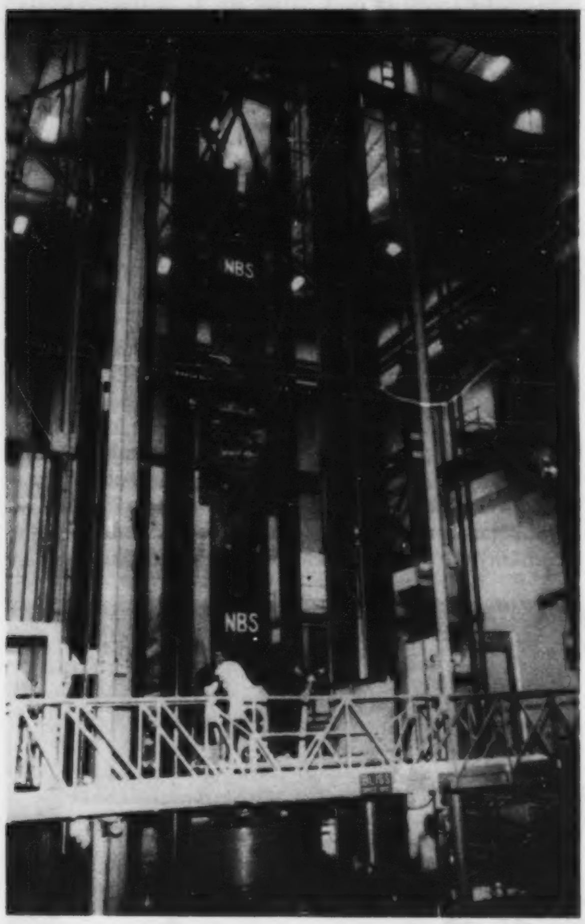
Piping

Environmentally Assisted Pipe Cracking. Cracks have been observed in the heat-affected zones of weldments in austenitic stainless steel piping in boiling water reactors (BWRs) since the mid-1960s. Since that time, indications of cracking have been found in all parts of the recirculation system, including large-diameter lines. (*See 1984 NRC Annual Report, p. 137.*)

Tests in 1985 have shown that at relatively slow strain rates some degree of transgranular stress corrosion cracking (TGSCC) occurs even at impurity levels corresponding to water conductivities of ×0.5+J mS/cm. Examination of specimens from interrupted tests showed that cracks occur early in the tests and confirmed that observed cracking is not a phenomenon brought about by the experiments. Pipe tests on Type 316NG SS in impurity environments are under way to investigate susceptibility under more prototypic loading conditions.


REGULATORY GUIDE
REGULATORY GUIDE
Reassessment of the Technical Bases for Estimating Source Terms
Draft Report for Comment
U.S. Nuclear Regulatory Commission
FOR COMMENT

NRC nuclear safety regulations and Regulatory Guides go through an exhaustive comment and approval process before they are published. These examples represent some of the steps in the process.



Wide-plate crack arrest tests on reactor pressure vessel steel specimens were performed during 1985 at the U.S. National Bureau of Standards (NBS) as part of an NRC-NBS joint effort. The specimens shown here are vertically mounted in the NBS-test machine (the largest available pressure loading machine in the United States), are deliberately flawed, subjected to several thermal gradients and then tested to produce long crack propagation and arrest.

A cooperative effort is in progress with the Electric Power Research Institute (EPRI) NDE Center to study the suitability of German "Nuclear Grade" Type 347 stainless steels for BWR primary piping systems. The work at the NDE Center will focus on the development and verification of welding practices for this material, and the work at the Argonne National Laboratory (ANL) will focus on the resistance of the material to stress corrosion cracking (SCC). Preliminary SCC tests have been carried out on one heat, or casting. It appears very resistant to both intergranular and transgranular cracking even in impurity environments. Additional heats have been obtained from Germany and are being welded at the NDE Center.

Additional data have been obtained to confirm the dependence of various SCC susceptibility parameters—such as the average crack growth rate and timeto-failure—on the applied (nominal) strain rate, and to test predictions of a phenomenological model based on estimates of crack tip strain rates obtained from elastic-plastic fracture mechanics. The model has been extended to obtain a relationship between average crack growth rate and the average crack tip strain rate. This relationship was found to be in good agreement with experimental results for intergranular and transgranular cracking of Types 304 and 316 SS in various environments containing dissolved oxygen and impurities, and it is consistent with a slip-dissolution model for crack advance.

The effect of impurity elements (sulfur and phosphorus) on grain boundary chromium depletion of Type 304 SS was also investigated. Analyses of scanning transmission electron microscopy showed that phosphorus strongly promotes chromium depletion at low temperatures, whereas sulfur does not.

Long-term fracture-mechanics-type crack growth tests were performed to quantify the influence of water chemistry (i.e., dissolved oxygen, hydrogen, sulfate concentration) on the rate and mode of crack growth of Type 304 SS. The results showed conclusively that crack growth in the sensitized materials ceased at the low dissolved oxygen levels even in the presence of sulfate in the water. The crack growth data from the fracture-mechanics-type specimens are consistent with the more extensive results from slow strain rate experiments concerning the effects of dissolved oxygen and sulfate on the SCC behavior of the material in high-temperature water.

An important aspect of the work this year has been the use of pipe and components removed from service and replaced with the new pipe and components. Materials received from the Hatch 2 plant (Ga.) are being used in metallurgical studies at ANL to help validate the remedies proposed by industry for the BWR pipe cracking problem and in NDE studies at PNL. Surface residual stress measurements have been made on an actual pipe-to-elbow weld overlay and a recirculation header-endcap overlay removed from the Hatch 2 reactor. The stresses on the inner surfaces of the weldments removed from service were compressive, although not as compressive as the stresses measured on mockup pipe-to-pipe prepared with nominally identical procedures. The stress distribution for the pipe-to-elbow overlay are strongly nonaxisymmetric in contrast to the mockup weldments. The stress distributions in the header-endcap weldment were close to axisymmetric, and the stress distributions and magnitudes are close to those obtained from the mockup weldments and axisymmetric finite element analyses.

A detailed metallographic examination of the recirculation header-endcap weld overlay from Hatch 2 was completed. The conclusions are similar to those obtained from the previous analysis of the pipe-to-elbow overlay. Blunting of crack tips was observed. There was no evidence of tearing or any through-wall extension of the crack beyond the blunted region. However, only relatively few short cracks were present in either weldment, although several of the cracks were quite deep (+J N60 percent throughwall). One interesting feature is that in both of the weldments the cracking occurred in the forged component (elbow or endcap), rather than in the pipe.

Irradiation-induced segregation of alloying elements or impurities in Type 304 SS can result in irradiation-assisted stress corrosion cracking (IASCC) of solution-annealed material in high-temperature water. The major environmental parameter that controls IASCC is the open-circuit corrosion potential. The extent to which hydrogen-water chemistry suppresses radiolysis and alters the open-circuit corrosion potential of core materials and the SCC behavior of the irradiated material has not been determined. Since in-reactor experiments to measure this effect are very complex, laboratory experiments are being performed to determine the effect of the intense gamma radiation on the open-circuit potential.

Piping Fracture Mechanics. NRC's ongoing programs in the piping fracture assessment area have been called upon to provide rapid evaluations of the rules and rule changes related to piping integrity analyses (see *1984 NRC Annual Report*, p. 138). Current research efforts are addressing the degree of conservatism in present piping analysis procedures by coupling analytical developments and refinements with pipe fracture experiments, using prototypical pipe specimens. The research efforts at the David Taylor Naval Ship Research and Development Center (DTNSRDC) at Annapolis, Md., continues to provide prompt response to technical issues arising from changes in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code rules for evaluating flaws in piping. Recent rule changes have provided guidance on evaluating flaws in welds and the DTNSRDC research has moved to validate those rules for nonflux welds.

The Degraded Piping II program at Battelle Columbus Laboratories constitutes the mainstay of the NRC's piping fracture research. The pipe test facility (see figure) is capable of testing large-diameter, heavy-wall piping at LWR operating temperature and loading conditions. The Degraded Piping II program provides a firm theoretical basis for piping fracture analyses, as well as the experimental data necessary to validate the analyses. This program has taken the lead in providing an assessment of the ASME Code rule changes for welds. In a cooperative effort between the NRC and EPRI, a series of flux welds were prepared in six-inch and 16-inch-diameter stainless steel pipe. Flaws have been introduced into those welds, and the pipes are being tested in the pipe test facility to determine the fracture resistance of flawed welds. Initial indications suggest that the rule changes are acceptably conservative.

Because of the expense involved in large-scale pipe tests, the NRC has been working with EPRI and several other countries in an effort to form a research consortium whose combined resources can provide the necessary research in a timely manner. Those efforts have produced the International Piping Integrity Research Group (IPIRG), whose objective is to develop, improve, and verify engineering methods for assessing the integrity of nuclear power plant piping containing defects, especially under severe dynamic and seismic loading. The NRC and contractor staff visited several European countries and Japan to solicit participation in the group. An organizational meeting is expected to be held in early 1986.

The close interaction among the contractors involved with piping research, coupled with the international cooperation being developed under IPIRG, will provide a firm technological base for establishing an international consensus on piping fracture behavior. Subsequent rulemaking should thus reflect known and accepted degrees of conservatism which will ensure safety on a technologically defensible basis.

Electrical and Mechanical Components

Nuclear Plant Aging Research. A multi-year, multidisciplinary program plan, NUREG-1144, was issued during the report period, which identifies the potential impact of component aging on safety, defines issues to be resolved, describes technical objectives of research and a research approach to achieve the program goals, and sets out proposed milestones and schedules.

A preliminary study was also completed on aging assessment and defect characterization of motor-operated valves (NUREG/CR-4234) and electric motors (NUREG/CR-4156). Measurements were performed at four operating plants to verify monitoring techniques and to obtain characteristic "signatures" indicative of degradation and misadjustments of motor-operated valves. This research supports the Office of Nuclear Reactor Regulation (NRR) in resolution of Generic Issue II.E.6.1, "In Situ Testing of Valves" (see Chapter 2).

A study was completed during the fiscal year to identify aging and service wear of hydraulic and mechanical snubbers used on safety-related piping and components of nuclear power plants. The ASME Section XI Code Committee and ANSI/ASME-OM4 Committee have been made aware of the results of the study. A value-impact analysis reflecting cost savings from reducing the number of snubbers in existing plants is being incorporated in a regulatory guide on qualification and acceptance tests for snubbers used in systems important to safety.

A series of tests was completed at the Shippingport Atomic Power Station (Pa.) to verify reflectometry techniques for detecting degradation in electrical circuits and to assess the status of selected electrical equipment before removal for additional testing and evaluation. This test program inaugurates an effort to obtain information from Shippingport on aging and service wear of electrical and mechanical components and primary coolant boundary materials, and on radionuclide inventory, for use in decommissioning evaluations.

Decommissioning. The NRC continued to develop an information base for use in decommissioning light-water reactors (LWRs) and other nuclear facilities, and published three reports during the year. The reports dealt with (1) decommissioning reference nuclear fuel cycle and non-fuel cycle facilities following postulated accidents (NUREG/CR-3293), (2) evaluation of nuclear facility decommissioning projects—Three Mile Island Unit 2 polar crane recovery (NUREG/CR-3884)—and (3) decommissioning LWRs at a multiple reactor station (Addendum to NUREG/CR-1755).

Proposed amendments to the regulations were published in February 1985, setting forth technical and financial criteria for decommissioning licensed facilities. The NRC sponsored an international conference in July 1985 on planning for nuclear reactor decommissioning.

A report summarizing all previously issued topical reports on the analysis of measurements of residual radionuclide contamination within and around commercial nuclear power plants will be published in 1986. Data needed to evaluate methods, radiation exposure, and costs of decommissioning nuclear facilities are still being collected. An annual summary report on the evaluation of nuclear facility decommissioning projects was published in 1985 (NUREG/CR-4090).

Spent Fuel Storage. Research was continued at the Idaho National Engineering Laboratory (INEL) on the effects of storing irradiated LWR fuel in a dry environment at low temperatures. Both defective and intact BWR and PWR assemblies stored in air and in non-oxidizing atmospheres were used. Two reports published during the year discussed (1) a dry spent fuel storage test plan for destructive fuel rod examinations (NUREG/CR-4084), and (2) the performance in inert gas and dry air storage atmospheres of spent LWR fuel rods that were deliberately made defective (NUREG/CR-4074).

The NRC is revising 10 CFR Part 72 to adapt it to the licensing of both short- and long-term storage of spent fuel and high-level waste in a monitored retrievable storage installation. These options for managing such materials were established in the Nuclear Waste Policy Act of 1982.

Nondestructive Examination

This program includes studies of improved methods for the detection and sizing of flaws during inservice inspection of carbon steel, wrought and cast stainless steel piping, and pressure vessels. It also includes studies of online continuous monitoring techniques (using acoustic emission) for crack growth and leak detection.

Flaw Inspection by Ultrasonic Methods. A vastly improved method for the detection and sizing of flaws in BWR stainless steel piping and other reactor primary components—called SAFT-UT (synthetic aperture focusing technique for ultrasonic testing)—is a computer-based testing procedure that produces highresolution, three-dimensional images of cracks and other material flaws. PNL is adapting this method for field inspections from earlier work done at the University of Michigan, where the technique was demonstrated in the laboratory. In 1985, a field system was assembled with its own computer and taken to a field site for a successful demonstration. Also a technique was developed along with the SAFT processing for optimum detection and characterization (sizing) of intergranular stress corrosion cracking in stainless steel piping. Finally, a special purpose SAFT processor prototype that will allow real-time processing of data and imaging of flaws on the spot, as the inspection is being conducted in the field, was developed. This allows for decisions to be made on the presence and nature of flaws in components while the inspection is being conducted.

Flaw Inspection of Centrifugally Cast Stainless Steel. Although the ASME Code requires that cast stainless steel piping be inspected, the current inspection techniques have not been fully satisfactory (see *1984 NRC Annual Report*, p. 139).

In 1985, an evaluation of various inspection techniques used worldwide was undertaken, in cooperation with the Nuclear Energy Agency of the Organization for Economic Cooperation and Development, to identify the most promising ones for the inspection of cast stainless steels. Data analysis from these inspections of flawed cast stainless steel specimens is under way, and improved procedures and techniques will be sought. Further work is planned to start in 1986 for a more thorough evaluation of the most promising techniques.

Inservice Inspection System Qualification. Research work and field experience over the last several years have indicated that inservice inspection as currently practiced is not always reliable. NRC research results have indicated a need for qualification of the entire inservice inspection process, including the personnel, procedures, and equipment. Research has been conducted and criteria developed for the proper qualification of the system. Subjects of greatest relevance for qualification are the education, experience, and examination requirements of inspection personnel; procedure requirements; equipment measurements; and evaluation and requirements for actual performance testing of the total personnel-procedure-equipment aggregation using actual components and realistic flaws, as a prerequisite to conducting an inspection on reactors. Based on the research conducted at PNL, criteria were prepared, reviewed (by NRC and the industry) and made final in 1985. These criteria formed the basis for NRC cooperation with the ASME in developing and implementing inservice inspection system qualification requirements in the ASME Code. Accordingly, in 1985, the NRC worked with designated Code committees to develop three documents for incorporation into the Code of recommended qualification requirements. At the close of the report period, these documents were being reviewed by the appropriate committees for acceptance into the Code.

Continuous Monitoring for Crack Growth and Leak Detection. Research and evaluation of leak monitoring systems were described in the *1984 NRC Annual Report*, pp. 139-140. Research work through 1985 at the Argonne National Laboratory (ANL) has produced techniques which use acoustic emission for accurate detection, location, and sizing of leaks from cracks in primary system piping. In 1985, the data base was extended to detect and quantify leaks as small as 0.001 gpm and up to one gpm from cracked piping, and techniques were developed and evaluated for accurate location of these leaks. Also, a field prototype system that could be used for actual plant monitoring was developed and tested. The techniques and equipment will be improved and validated by conducting on-line reactor monitoring in 1986 and 1987.

Research has also been under way at PNL using acoustic emission for the continuous on-line monitoring of reactors to detect and locate crack growth and to estimate the severity of the cracking from the acoustic emission signals. Up to 1985, a large body of laboratory and some field data have been developed to establish feasibility and methodology for inservice monitoring of reactors and for evaluation of the data. In 1985, a great deal of data from an intermediate-scale test using a pressure vessel—which was conducted over a one-year period in the Federal Republic of Germany and which produced crack growth under simulated reactor operating conditions—was thoroughly evaluated to upgrade and validate existing models and technology. The evaluation established that the continuous monitoring of reactors for crack growth is feasible. This technology will be validated by actual monitoring of an operating reactor (TVA's Watts Bar Unit 1 in Tennessee) during 1986 and 1987. The availability and proper use of this technology will mean that reactors can be continuously monitored and that any cracks that develop can be continuously detected and evaluated. In this way, proper and timely action can be taken to avoid extensive crack growth so that leakage or failure would never occur.

EQUIPMENT QUALIFICATION

Qualification of Electrical Equipment For Harsh Environments

During fiscal year 1985, a new research program was initiated, in cooperation with the French, to compare the relative effect of beta and gamma radiation on the mechanical and thermal behavior of polymers during an accident. (The total beta energy emission from fission products released to containment during an accident is an order of magnitude greater than that for gamma emissions.) This research will assess the damaging effects of the beta radiation, which has a relatively short range in most materials. The results of the NRC source term research study (NUREG-0956) are to be used in conjunction with the beta radiation effects research to evaluate the adequacy of the accident radiation dose calculational assumptions and models contained in Revision 1 of Regulatory Guide 1.89 on the environmental qualification of electrical equipment important to safety for nuclear plants.

Extensive modifications were made during the year to the high-intensity adjustable cobalt array facility (HIACA) at Sandia National Laboratories in order to provide for superheated steam and air overpressure capability and thereby better simulate a range of accident conditions, from design basis loss-of-coolant to severe accidents. The accident simulation is used in research on qualification methods for and survival of electrical equipment. Tests have been started in the HIACA using Class 1E ethylene propylene rubber insulated cables to compare the sequential steam and radiation accident simulation procedures for qualifying cables with saturated steam with the results from using simultaneous superheated steam and radiation, which more closely simulate anticipated accident conditions.

A detailed technical understanding of the domain and phenomena of both homogeneous and heterogeneous oxidation in the aging degradation of polymers was developed and reported in NUREG/CR-4008. This research further extended knowledge of dose-rate effects in this area.

Fire Protection Research. The 1985 fire protection research concentrated on the characterizing of potential fires, the development of an analytical technique for the prediction of resultant conditions in various plant areas, and the determination of failure thresholds of safety-related components under such conditions. Results are being applied to the assessment of probabilistic fire risks in critical plant areas, such as the control room. Tests were conducted in 1985 to determine the heat and effluent releases associated with a spectrum of fires in electrical cabinets typical of nuclear power plant control rooms. A computer simulation was used to reproduce conditions produced by such fires in a typical control room. Various safety-related components have been tested to determine their failure thresholds in fire and fire-suppressant environments. The ongoing program will continue in 1986 with the component failure threshold testing and a series of fire tests in partial replications of control rooms to verify the computational results.

Batteries. Seismic fragility tests of 10-year-old Exide and LCU battery cells removed from the Calvert Cliffs (Md.) and North Anna (Va.) nuclear stations were described in NUREG/CRs-4095 and -4096. These tests found that Exide- and LCU-manufactured lead calcium cells survived and functioned in a seismic test response spectrum well in excess of that used in the design of most nuclear plants, as did the previously tested Gould cells from the FitzPatrick plant (N.Y.).

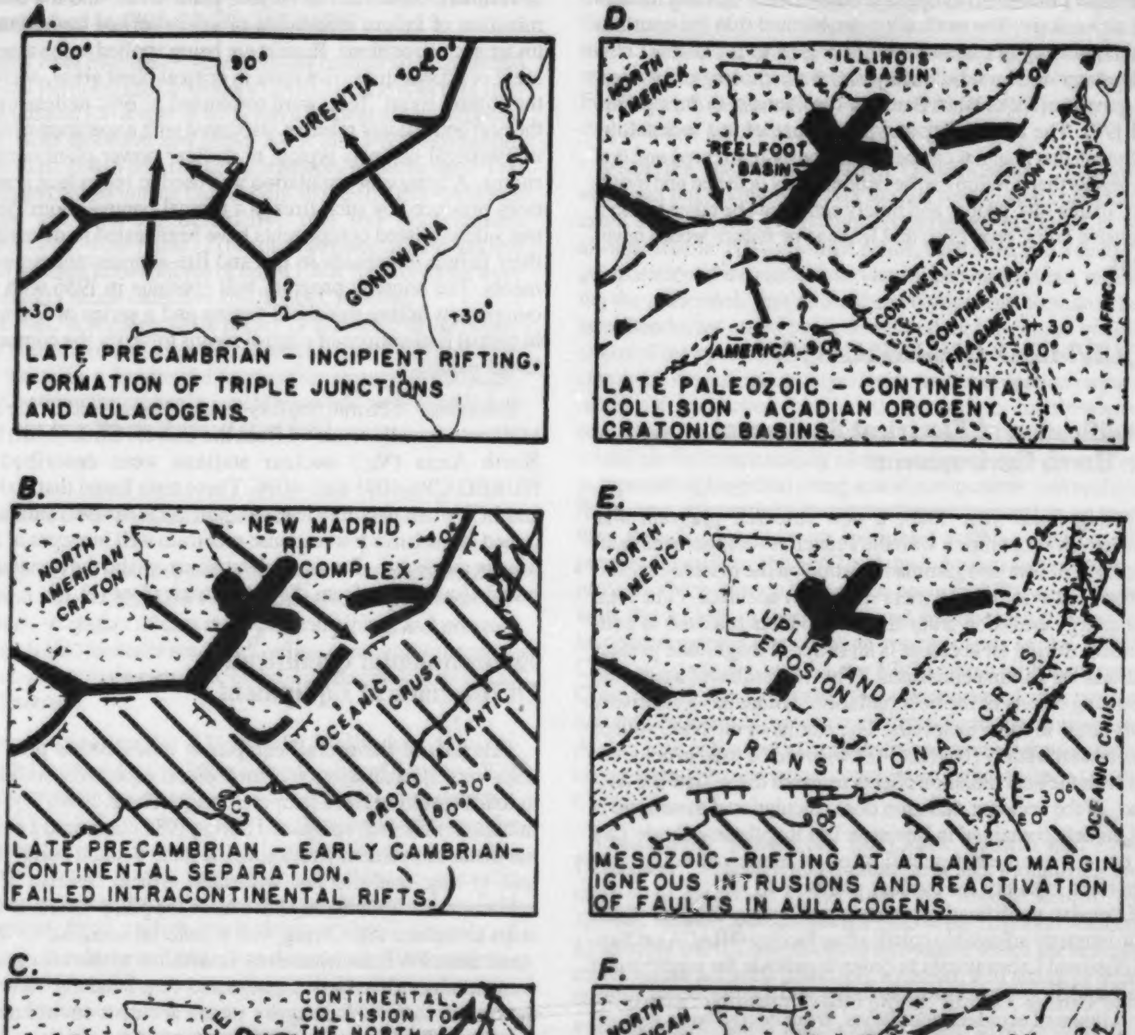
Environmental Qualification Of Mechanical Equipment

This part of the overall equipment qualification program deals with the qualification of mechanical equipment subjected to loads resulting from temperature, pressure, humidity, and radiation. Research completed during 1985 uncovered a potential material problem with a specific main coolant pump shaft seal O-ring that did not withstand typical internal PWR environments. This finding has caused the pump vendor to take steps to replace this O-ring with a material adequate to withstand those PWR environments. In addition to this effort, tests were conducted on typical purge and vent butterfly valves to determine their leak integrity under accident environments involving rising containment pressure and temperature. The integrity of the elastomeric seal and packing in these kinds of butterfly valves is affected by rising pressure and temperature.

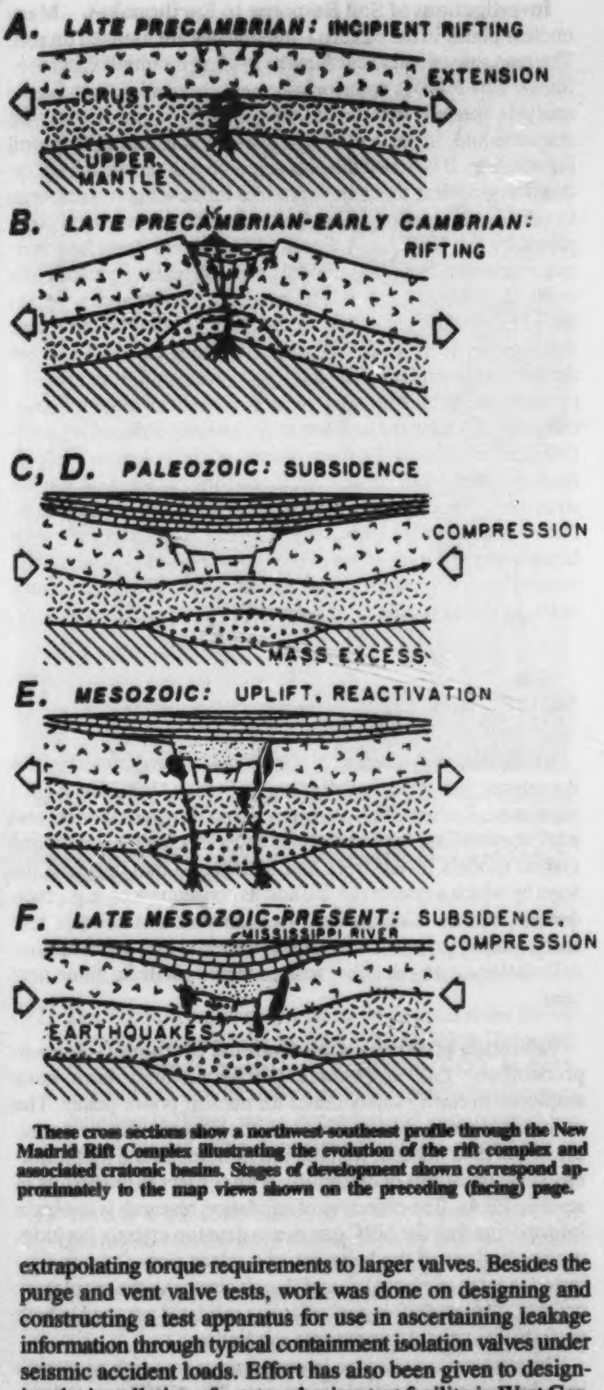
All the above results provided NRC licensing staff with the technical basis for evaluating the integrity of mechanical components under accident conditions.

Dynamic Qualification of Equipment

This part of the overall equipment qualification program deals with the dynamic (including seismic) qualification of mechanical and electrical equipment. Tests completed in 1985 validated the NRC licensing practice for estimating torque motor requirements for closing typical purge and vent butterfly valves against various inlet flow conditions. The data obtained during these tests have also provided a basis for


A.
100°
90°
LAURENTIA
40°
20°
GONDWANA
?
+30°
+30°
LATE PRECAMBRIAN – INCIPIENT RIFTING, FORMATION OF TRIPLE JUNCTIONS AND AULACOGENS.
B.
NEW MADRID RIFT COMPLEX
NORTH AMERICAN CRATON
40°
OCEANIC CRUST
PROTO ATLANTIC
30°
80°
LATE PRECAMBRIAN – EARLY CAMBRIAN– CONTINENTAL SEPARATION, FAILED INTRACONTINENTAL RIFTS.
C.
CONTINENTAL COLLISION TO THE NORTH
SUBSIDENCE
TROUGH
OUACHITA
SUBDUCTION
PROTO ATLANTIC
?
EARLY TO MIDDLE PALEOZOIC – TACONIC OROGENY, SEDIMENTATION IN CRATONIC BASINS.
D.
NORTH AMERICA
ILLINOIS BASIN
REELFOOT BASIN
CONTINENTAL COLLISION
CONTINENTAL FRAGMENT
AMERICA
90°
30°
80°
AFRICA
LATE PALEOZOIC – CONTINENTAL COLLISION, ACADIAN OROGENY, CRATONIC BASINS.
E.
NORTH AMERICA
UPLIFT AND EROSION
TRANSITIONAL CRUST
OCEANIC CRUST
?
30°
80°
MESOZOIC – RIFTING AT ATLANTIC MARGIN, IGNEOUS INTRUSIONS AND REACTIVATION OF FAULTS IN AULACOGENS.
F.
NORTH AMERICAN CRATON
MISSISSIPPI EMBAYMENT
UPLIFT
TRANSITIONAL CRUST
OCEANIC CRUST
CRETACEOUS TO PRESENT EMBAYMENT AND COASTAL SUBSIDENCE. COMPRESSIONAL STRESS REACTIVATES FAULTS.

These schematic diagrams illustrate the plate reconstruction of the North American craton and interactions with adjacent plates and geologic activity of the New Madrid Rift Complex during the last 600 million years. The outline of the State of Missouri is shown for location and approximate scale. (The term "craton" denotes a stable, relatively immobile area of the Earth's crust that forms the nuclear mass of a continent.)


A. LATE PRECAMBRIAN: INCIPIENT RIFTING
EXTENSION
CRUST
UPPER MANTLE
B. LATE PRECAMBRIAN-EARLY CAMBRIAN: RIFTING
C, D. PALEOZOIC: SUBSIDENCE
COMPRESSION
MASS EXCESS
E. MESOZOIC: UPLIFT, REACTIVATION
F. LATE MESOZOIC-PRESENT: SUBSIDENCE, COMPRESSION
MISSISSIPPI RIVER
EARTHQUAKES

These cross sections show a northwest-southeast profile through the New Madrid Rift Complex illustrating the evolution of the rift complex and associated cratonic basins. Stages of development shown correspond approximately to the map views shown on the preceding (facing) page.

extrapolating torque requirements to larger valves. Besides the purge and vent valve tests, work was done on designing and constructing a test apparatus for use in ascertaining leakage information through typical containment isolation valves under seismic accident loads. Effort has also been given to designing the installation of a gate valve in a test facility in West Germany to determine the characteristics of flow-induced forces that can be developed during valve operations.

It is intended that these test results will provide NRC licensing with a basis for evaluating the integrity of valves when subjected to seismic- and flow-induced loads.

SEISMIC RESEARCH

Seismic Hazard

The NRC research program in seismology and geology continued to concentrate on seeking better definition of seismic hazards in the United States east of the Rocky Mountains. The program is directed toward quantifying these hazards, using probabilistic techniques where appropriate. Three aspects of the subject that contribute significantly to the uncertainty in seismic hazard estimation are seismic zonation, attenuation of seismic waves, and site-specific response. Seismographic networks and geological/geophysical studies are used to address these items and to define relationships between crustal features and deep-seated tectonics. The NRC continued support for the established seismographic networks by replacing older station equipment with digital instruments and by deploying additional strog-motion seismographs.

Tectonic Investigations in Oklahoma. Recent studies in Oklahoma have disclosed what is probably the first capable fault in the United States east of the Rocky Mountains with surface exposure. The Meers fault, located in southwestern Oklahoma, has been found to show signs of geologically recent movement. Although the existence of the fault has been known for a long time, it was previously assumed to be of Paleozoic age, like many of the faults in the midcontinent area. The NRC-supported studies provided definitive data on the age of the fault's most recent movement. These studies included mapping, trenching, age-dating of organic materials, and low sun-angle photography. Seismographs placed near the Meers fault have not, so far, registered any seismicity on the fault. Investigations of the Meers fault during 1985 have proved with a fair degree of certainty that movement of the fault occurred in recent geologic time, most probably 1,100 years ago. This determination is based on stratigraphic studies, including age-dating and mapping of two trenches across the fault. Further, refinement of the age of faulting will result from C-14 dating of samples from the trenches. Trench-mapping indicates that the fault movement at the surface was a high-angle reverse slip with no strong indications of strike-slip displacement. These findings give a new perspective to studies of seismic hazard in the midcontinent region. Among the questions to be resolved in the future are the possibility of recurrent movement on the fault, whether or not there was substantial strike-slip movement, and what other faults may be found in the midcontinent that are active and have surface expression.

Tectonic Investigations in South Carolina. In November 1982, the U.S. Geological Survey (USGS) noted that the 1886 Charleston earthquake could not be associated with a known geological structure. This implies that there is a probability, however small, that the level of ground motion associated with a Charleston-sized earthquake could occur elsewhere on the eastern seaboard. More recently, high seismic accelerations were recorded at sites in New Brunswick, Canada, and New Hampshire and Arkansas. A special study of the Charleston, S.C., area was started this year with the goal of testing hypotheses concerning the Charleston earthquake of 1886. The study is using geophysical and geological data together with topography and satellite imagery to devise a model of crustal stress distribution. The stress model and other data will test the various hypotheses that have been advanced to explain the earthquake. It is expected that a clearer picture of the seismicity will be formed by eliminating hypotheses that are not consistent with prevalent stress directions and other data. Other stratigraphic, seismological and geophysical studies have provided further details on the crustal structure near Charleston. These studies have defined a trough on the coast that is bounded by faults and appears to be a region of continuing subsidence.

Tectonic Investigations in Tennessee. Further west, in Tennessee, studies of seismicity and crustal structures have shown that a rift-type structure is the most likely cause of seismic activity in central Tennessee and the adjacent parts of Alabama. The conclusion is based on an analysis of gravity, magnetic, and seismic network data.

New Madrid Seismotectonic Program. In 1985, the cooperative New Madrid seismotectonic program was completed. That program had been in existence since 1976, and it integrated geologic, geophysical and seismic investigations within a 200-mile radius of New Madrid, Mo. The States of Alabama, Arkansas, Illinois, Indiana, Kentucky, Mississippi, Missouri, and Tennessee, and the USGS, the Tennessee Valley Authority, and the NRC all participated in the study. The purpose of the program was to determine (1) the causes of the three large earthquakes that occurred near New Madrid in the winter of 1811-1812 and the 22 damaging earthquakes that have been felt in the area since then, and (2) the earthquake ground motion parameters characteristic of this region. The 1811-1812 earthquakes are noteworthy because of their great intensity (which was estimated at Modified Mercalli Intensity XII), because of the large area affected (they were felt as far away as Charleston, S.C., Washington, D.C., and Hartford, Conn.) and because of their location in the interior of the North American Continental Plate (in contrast, most large earthquakes occur at the edges of large plates that make up the earth's crust). Integration of the results from the New Madrid seismotectonic program has produced confirmatory information in support of a conceptual model which posits that contemporary seismicity occurs along geologic structures produced during or following an episode of crustal rifting that took place one billion years ago. Subsequent to rifting, the area subsided, was uplifted, experienced the emplacement of large igneous plutons, and subsided. The current stress regime is compressional, in contrast to the tensional stress field that produced the initial rifting and accompanied the emplacement of the igneous plutons. Nevertheless, the current stress regime has reactivated pre-existing geologic structures. The net result of this NRC-funded research is that, for estimates of seismic hazard, the present state of knowledge indicates that the seismically hazardous region around New Madrid is confined to a well-defined zone, which runs roughly parallel to the Mississippi River from northern Arkansas northeastward through southeastern Missouri and eastern Tennessee.

Investigations of Soil Response to Earthquakes. Many nuclear plants in the eastern United States are founded on soil. The response of soil to earthquake loading is complex and nonlinear. The NRC is sponsoring research to validate dynamic analysis models that would be capable of predicting soil response and, in particular, soil settlement resulting from soil liquefaction. It is hoped that this research will reduce the uncertainty in modeling the major variables influencing soil response to earthquake loading. The first phase of the research completed by the U.S. Army Corps of Engineers describes various currently available models for evaluating seismically induced settlement in soil. The results of the research program have been published as NUREG/CR-3380. One of the major findings of the study is that only a nonlinear, effective-stress model that considers porewater pressure generation and dissipation, material softening, reduction in shear modulus, etc., can rationally treat the problem of seismically induced liquefaction and settlement. In later phases of the research project, models of embedded structures, simulating nuclear power plant structures, were tested in special centrifuge experiments conducted at Cambridge University, England. The test results were being analyzed at the close of the report period to evaluate the capabilities of the several models identified in the first phase of the study to predict soil behavior.

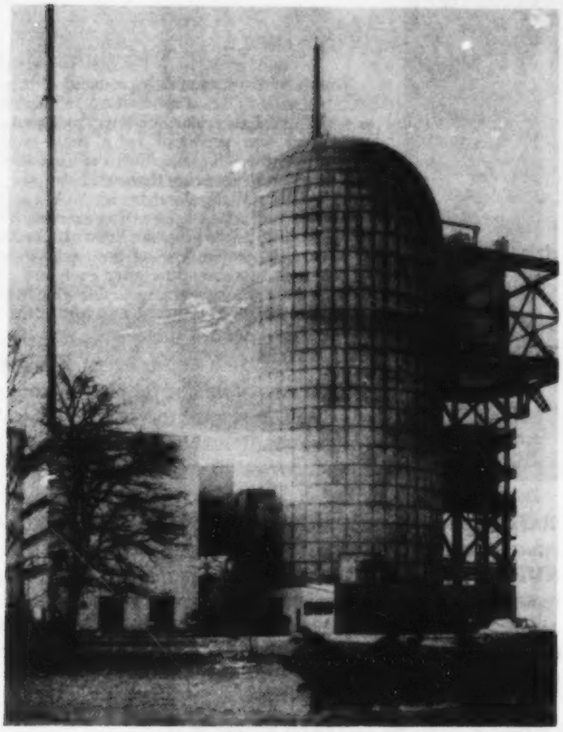
Seismic Risk

BWR Risk Assessment. The research program to assess the seismic risk of the LaSalle County Station Unit 2 BWR (Ill.) continued. During 1985, estimates of building, component and equipment failure have been made and are being integrated into system models (event trees and fault trees) that describe the ways by which a system can fail and its consequences, e.g., radioactive release. Calculations to determine the seismic risk using simplified system models began late this year. Similar calculations using detailed system models will be made next year.

Validation of Seismic Calculational Methods. Seismic probabilistic risk assessment (PRA) methods have been employed to clarify safety issues for nuclear power plants. The randomness of the seismic hazard, the uncertainties and variety of the data needed, and the inexactitude of the methodology raise questions of credibility with respect to the results of seismic PRAs. The objective of validation research is to obtain information that the NRC can use to develop criteria for judging predictions of the behavior of nuclear power plants subjected to large earthquakes and thereby improve the regulatory process. The predictive methods to be validated are used in both probabilistic and deterministic predictions.

The strategy is to engage in cooperative research programs in order to stretch available resources. The NRC is participating in the following three efforts:

(1) A soil-structure interaction experiment being performed in Taiwan by the Electric Power Research Institute (EPRI). Initial efforts have focused on construction of



The HDR facility, located on the Main River at Kahl, West Germany, is a decommissioned power plant that was modified for the performance of thermal-hydraulics and materials engineering experiments. A large mechanical shaker will be used on the containment building in earthquake-related experiments in 1986.

a test structure and low-level tests with a mechanical shaker. Measurements of response during earthquakes will be made and compared with predictions.

(2) The Phase II experiments being performed at the Heissdampfreaktor (HDR) facility in Kahl, West Germany, by Kernforschungszentrum Karlsruhe (KfK). In 1985, a large mechanical shaker, capable of exciting the containment building, was constructed and a test piping loop was designed. The experiments will be carried out in 1986.

(3) Tests of a 1/2-scale model of a PWR piping loop to be performed on the large shaker table in Tadotsu, Japan, by the Nuclear Power Engineering Test Center (NUPEC). In 1985, modifications to the Japanese test specimens were designed that would permit excitation of the specimen well into the inelastic range. The experiment is planned for 1987.

Standard Problems for Structural Computer Codes. An effort started in 1984, which included a task to determine the limitations and applicability related to the soil-structure-interaction methods used by industry, was completed this year. The results of this task, which are based on experimental and actual earthquake data, are discussed in NUREG/CR-4182.

Seismic Margins

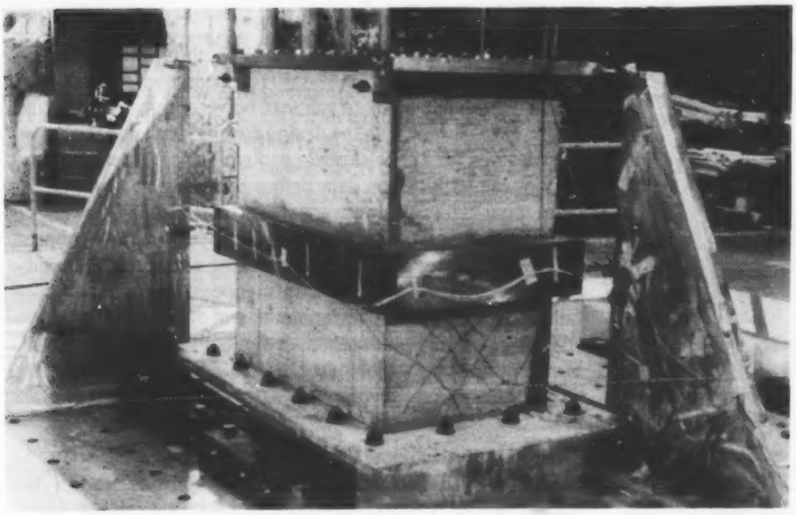
Seismic Design Margins. In 1985, the Expert Panel on Quantification of Seismic Margins pioneered an approach for assessing the adequacy of nuclear power plants to withstand earthquakes larger than the design bases. The panel evaluated the findings of recent seismic probabilistic risk studies and earthquake experience data before outlining procedures for plant seismic margins reviews. (NUREG/CR-4334 was issued explaining the general approach.) These procedures are based on a systems screening approach and the concept of the establishment of a level of "high confidence of low probability of failure" for important components and systems.

The panel interacted extensively with the internal NRC Working Group on Seismic Design Margins in developing these procedures for seismic margins reviews. Trial plant reviews using these guidelines will be conducted in 1986.

Seismic Component Fragilities. This new endeavor seeks to test the hypothesis that component fragilities used in past seismic PRAs underestimate component capacities and have led to unreasonable concerns about the earthquake threat to nuclear power plants. Reliance is given mainly to data acquired from utilities, manufacturers and private testing laboratories, and their interpretation, but some limited testing is also foreseen. During this year, lists of critical components were prepared with help from industry, and test data were collected and evaluated. A scheme for ranking and grouping components for fragility testing was developed. Based on this scheme, program emphasis will be on electrical components, which are considered the major risk contributors, though mechanical components are also covered.

Seismic Category I Structures Program. The static and dynamic testing of reinforced concrete models representing portions of nuclear power plant noncontainment buildings (i.e., wall and floor segments) continued this year. This current series of tests will continue through 1986 to investigate the large differences observed when analytical predictions of building responses are compared with experimental data. Since little data are available on models of nuclear power plant buildings, it is not known if these analytical-experimental differences are principally due to the models tested, interpretation of test data obtained, or shortcomings in standard analytical procedures used. The overall goal of this program is to assess the ability of Category I structures other than the containment to sustain earthquake motions in excess of their original design bases.

Load Combinations for Piping Systems. The probability of pipe rupture in the primary coolant loop of equipment made by each American PWR reactor vendor has been calculated by investigating both fatigue crack growth and seismically induced heavy component support failures, as the mechanisms for causing pipe rupture. Results indicate that pipe ruptures are extremely unlikely, and this conclusion has led to a modification of General Design Criterion 4. The rule will permit the elimination of unnecessary hardware, such as pipe whip restraints and jet impingement barriers in PWR primary coolant loops. Radiation exposures averted because of reduced inspection and maintenance requirements are estimated at 35,000



The photo shows seismically-produced damage in a 1/10-scale model of a two-story, reinforced concrete diesel generator building. Earthquake simulation was produced by a shake-table operated by the U.S. Army Corps of Engineers Construction Engineering Research Laboratory at Champaign, Ill. Steel weights on the first and top floors were added to ensure an exact representation of an actual generator building. Cracks visible on the bottom level of the model were traced with marking pens after each test to monitor their subsequent growth following tests of greater severity.

person-rems; associated industry cost savings are thought to exceed $200 million. Similar work on BWRs is yielding substantially higher pipe rupture probabilities attributable to intergranular stress corrosion cracking mechanisms. BWRs are not yet covered by any rulemaking actions.

Pipe Damping Studies. A data base on pipe damping using input from domestic and foreign sources was established this year. This expanded set of data was used to confirm new damping criteria developed by the Pressure Vessel Research Committee and later endorsed by the American Society of Mechanical Engineers (ASME) and the NRC Piping Review Committee.

Dynamic testing of a representative three-dimensional piping system began this year. This testing will determine the sensitivity of pipe damping values to various design parameters such as force input, insulation, pressure, and support configuration. Early results indicate that the latter design feature has the greatest effect on damping. Testing will continue next year with the additional consideration of damping for high-frequency (non-seismic) modal response.

Multiple Response Spectra Method Techniques. Analytic studies have provided the bases for an NRC position on "multiple response spectra method" techniques. The Independent Support Motion (ISM) method, which is based on a systematic evaluation of the response margins relative to time-history analysis, was accepted by the licensing staff. The ISM method may provide a basis for further reducing the number of piping supports used in nuclear power plants.

Probability-Based Load Combinations for Nuclear Structures. During the report year a method for combining loads (e.g., earthquake and accident pressure) that incorporated the probability of their occurrence during the nuclear power plant's operating life was developed. This method will provide a uniform safety margin when designing structures for different combinations of loads. Recommendations regarding the probability-based load combination criteria for designing concrete containment structures were published this year in NUREG/CR-3876. The draft of a similar report containing recommendations on the design of concrete shear walls was completed and issued for comment along with draft reports on tangential shear in concrete containments and reliability analysis for concrete shear walls.

Other External Hazard Research

Severe Weather. Severe weather research in 1985 was primarily concerned with damage surveys of areas struck by tornadoes. The purpose of these damage surveys is to obtain information on the length and width of the tornado damage path and the severity of the tornado. Compilations of these data are needed to better define the risk to nuclear facilities posed by tornadoes. The widespread tornado outbreak of May 31, 1985, in Ohio and Pennsylvania and in Ontario, Canada, was among those investigated this year.

Surface-Water Hydrology. A study of models used in the evaluation of ultimate heat sinks for nuclear power plants was completed. A technical report of the study, which provides information on the selection of appropriate models for specific safety design applications, was published as NUREG/CR-4120 in March 1985.

Meteorological and wave-surge measurements related to hurricanes were collected by an instrument network along the Florida coast. This information will help improve models that predict flood levels during hurricanes, so as to ensure that nuclear power plants in coastal areas are designed to withstand hurricane-induced flooding.

Ground Water Hydrology. Studies were completed on methods to control or reduce the contamination of ground water in the event of a nuclear reactor accident. This study evaluates protective strategies for retarding the movement of the contaminants; it is described in NUREG/CR-4251, issued in August 1985.

REACTOR OPERATIONS AND RISK

Risk Analysis

Risk Assessment Methods Development. The Risk Methods Integration and Evaluation Program (RMIEP) was initiated in 1984 to develop improved assessment methods to support probabilistic risk assessments (PRAs) of nuclear power plants. Work in 1985 concentrated on incorporating newly developed methods dealing with common cause failures, such as fires and human error, and also external events, such as earthquakes and floods, into current PRA techniques.

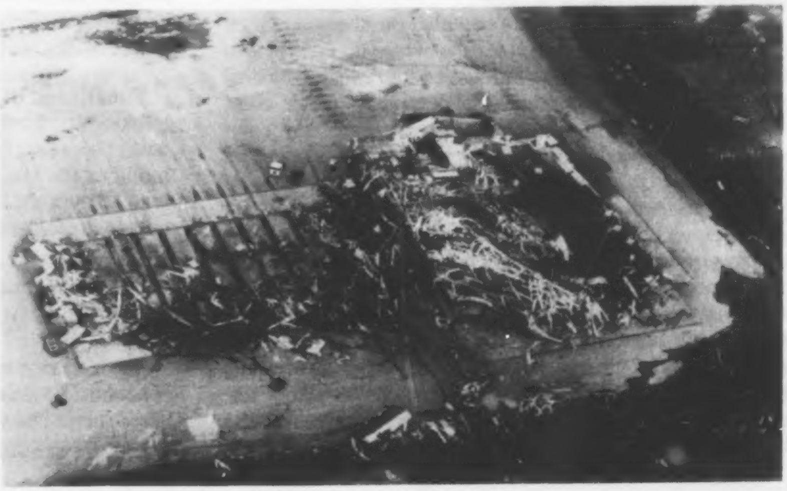
Integrated logic models for the LaSalle County Station Unit 2 (Ill.) nuclear plant safety systems—which include internal events, external events and dependent failures—are nearing completion. Draft reports on dependent failure analysis and uncertainty analyses were issued in 1985. The NRC continued to collect and analyze data from selected power plants, including failure data reports on valves, pumps, diesel generators, batteries, and inverters. These data are currently used in support of RMIEP and the Reference Plant Study. Planning efforts continued on the development of a comprehensive, agency-wide data acquisition system. These planning activities and data collection programs are being consolidated in the Office of Analysis and Evaluation of Operational Data (AEOD—see Chapter 4). This process will be completed in 1987.

Methods Development for Risk Reduction. As part of the Severe Accident Risk Reduction Program, the NRC is evaluating risks associated with six nuclear power plants: Surry(Va.), Peach Bottom (Pa.), Sequoyah (Tenn.), Grand Gulf (Miss.), LaSalle (Ill.) and Zion (Ill.). These studies are being carried out principally at Sandia National Laboratories with support from the Idaho National Engineering Laboratory and Battelle Columbus Laboratories. Each of the six plants was selected to represent a major type of containment design, i.e., BWR Mark I, II, and III and PWR large dry, ice-condenser and subatmospheric types. The risk reduction produced by various accident prevention and mitigation alternatives, such as filtered-vented containment vessels, is presently being evaluated for each facility. This program has been integrated with the Accident Sequence Evaluation Program (ASEP), described below, in order to provide a full risk profile (not including external events) for the six plants. The results of this evaluation will be used to prepare a comprehensive report (NUREG-1150) that will provide the basis for future regulatory actions based on PRA insights.

Human Factors

Human Reliability. This program provides the data necessary to support licensing evaluations of (1) nuclear power plant reliability programs that deal with the performance of plant personnel, and (2) the resolution of generic safety issues through the use of pertinent data from the human element portions of reliability assurance programs. The products of this research are also being applied toward resolving issues raised in the TMI Action Plan (NUREG-0660) and the Human Factors Program Plan (NUREG-0985, Revision 1). Major products of 1985 research included methods for acquiring probabilistic data on human reliability for use in reactor-safety PRAs. These methods involve the use of both expert judgment and computer modeling. Specifications for a human reliability data bank and a model for quantifying error probabilities associated with human action sequences were also developed. Procedures for systematically using PRA data and results in order to identify plant retrofit requirements, to examine retrofit alternatives, and to identify immediate and long-term human reliability research needs were completed. Eight publications reporting the research carried out under this program were issued during 1985.

Organization and Staffing. Objective, safety-related performance appraisals are essential to NRC assessments of the organizational effectiveness of utilities operating nuclear power plants; they are also needed for establishing staffing require-



The NRC Severe Weather Research Program in 1985 concentrated on damage surveys of areas hit by tornadoes. This photo shows what is left of a trucking plant in Wheatland, Pa., following a tornado outbreak of May 31, 1985, covering wide areas of Ohio, Pennsylvania and Ontario, Canada.

ments to ensure safe plant operation. This research aids in the resolution of organizational and staffing issues raised in the TMI Action and Human Factors Program Plans. Major products of 1985 research included an initial set of performance measures, with validity indices, for assessing organizational effectiveness at operating plants and data for comparing alternative control room staffing plans during normal and abnormal conditions. Two publications reporting research completed under this program were issued during 1985. Funding for this research program was terminated at the end of fiscal year 1985.

Operational Readiness. In order to better ensure the safe and timely response of plant personnel to normal and abnormal plant conditions, research was carried out in 1985 related to the upgrading of NRC operator qualifications, to operator training and licensing requirements, and to the assessment of nuclear power plant operating procedures. This research produced a computerized task analysis profiling system for identification of entry-level operating personnel skill, knowledge, ability and attitude; a technical basis for evaluating nuclear power plant simulation facilities for use in the NRC operator licensing process; a method for systematic assessment of the advantages and disadvantages of different formats and media for presenting nuclear power plant operating procedures; an assessment of the impact of upgrading emergency operating procedures on the NRC task analysis data base; and the application of the NRC task analysis data base to catalogue control room skills and knowledge, together with a method for using the catalogue for assessing personnel qualifications. Seven publications reporting research under this program were issued during 1985. Funding for this program was terminated at the end of fiscal year 1985.

Man-Machine Interface. The interactions between operators and the systems they control is an important consideration in ensuring the safe operation of nuclear power plants. In 1985, research was carried out to provide data needed for a sound technical basis by which to evaluate man-machine relationships and information exchanges in control rooms or other control areas. Research was conducted to assess and recommend human factor-based standards and guidelines for new or improved control system designs.

Research in 1985 produced a compilation of human factor guidelines for evaluating and assessing new or improved video display designs that may be introduced into existing control rooms. Laboratory experiments and data analysis were completed on the effects of psychological stress on operator decisionmaking. Procedures, operator training and diagnostic equipment used in analysis and response to seismic events were surveyed as part of the stress evaluation program at six nuclear power plants near seismically sensitive zones. A multiyear experimental program was completed to determine the effectiveness of artificial intelligence in fault diagnosis. A feasibility analysis was performed to identify probable sources of human factor-related problems that may be reducing the effectiveness of ultrasonic inservice inspection. A testing plan was developed for a limited field test in a nuclear power plant training simulator of an alarm reduction method, in order to assess the performance of control room operators under conditions of reduced annunciator alarms. Eight reports were issued under this program in 1985. Funding for this research program was terminated at the end of fiscal year 1985.

Accident Management. Accident management research seeks to identify specific operator actions that could and should be taken to mitigate the consequences of severe reactor accidents. This research contributes to the technical basis for the Commission's policy statement on severe reactor accidents. An accident management methodology was developed and a pilot plant application was initiated during 1985. This research project is aimed at developing methods for evaluating the effectiveness of operating plant personnel actions and emergency operating procedure guidelines and to identify the kinds of equipment modifications that could help mitigate the effects of a severe accident. Reviews of severe reactor containment failure events to identify the potential consequences of two major types of severe accidents were completed. These reviews have provided a technical foundation for the development of the accident management methodology. A review of existing research results, NRC regulations related to accident management, and industry accident response programs was also completed. A comprehensive report was issued in 1985 that documents the technical and programmatic research plan for future work in this area.

Management of NRC Rulemaking

Control of Rulemaking. In February 1984, the NRC Executive Director for Operations (EDO) directed that all offices reporting to the EDO and responsible for rulemaking must obtain the EDO's approval to begin and continue a specific proposed rulemaking action. The reason for the directive was to ensure that rulemakings were genuinely necessary and would be effective, efficient, timely and of high quality.

The EDO directed that RES independently review prospective rulemakings and make recommendations to the EDO as to whether to proceed with them. During 1985, RES completed 72 initial reviews of the 81 rulemakings sponsored by EDO offices. Late in the year, RES began a long-range program to conduct annual independent reviews and make independent recommendations to the EDO concerning ongoing rulemakings.

Timeliness of Rulemaking. RES also established a tracking and feedback system to help EDO ensure the timeliness of approved rulemaking actions. To accomplish this, RES modified the existing updating of entries in the NRS Regulatory Agenda (NUREG-0936) to require a timetable for each ongoing rulemaking sponsored by an office reporting to the EDO.

Emergency Preparedness

A rule change related to the potential complicating effects of earthquakes on emergency planning was published as a proposed rule in December 1984. A final rule was expected by late 1985.

A proposed rule on emergency preparedness for fuel cycle and other radioactive material licensees was prepared for Commission review. It was expected to be published for public comment by the end of 1985 with the final rule scheduled for 1987.

Atmospheric Dispersion. Work has continued on completing the analysis and documentation of meteorological and tracer data collected during previous field tests (see *1983 NRC Annual Report*, p. 127) conducted to evaluate atmospheric dispersion models. This research is being done to identify those models that are capable of and suitable for real-time predictions of the atmospheric transport and diffusion of effluents through the airborne pathway during and immediately following an accidental release of radioactive material from a nuclear power plant. Volume 3 of NUREG/CR-3488 (February 1985) gives information on the 1981 Idaho field experiment. Other technical reports published this year related to research in atmospheric dispersion are NUREG/CR-4072 (January 1985), which addresses the estimation of atmospheric dispersion at nuclear power plants using real-time anemometer statistics; NUREG/CR-4157 (March 1985), which provides a scientific critique of available models for real-time simulations of dispersion; and NUREG/CR-4158 (April 1985), which provides a compilation of information on the uncertainties involved in deposition modeling.

Research continued at the Idaho National Engineering Laboratory (INEL) to measure the washout and wet depositive factors for the chemical forms of airborne radioiodines released to the environment during periods of precipitation and fog. A study of the comparison of the 1981 INEL dispersion data with results from a number of different models (NUREG/CR-4159) was published in May 1985.

Fuel Cycle Risk Analysis

In June 1985, the NRC published a report on the potential consequences of accidents at fuel cycle and other radioactive material facilities. The report, NUREG-1140, considered accidents at 15 types of facilities. The most serious accidents were identified as fires and uranium hexafluoride releases. The significance of these accidents with regard to emergency preparedness was discussed.

Transportation Safety Research

Efforts were made in 1985 to document the extent of protection provided by spent fuel casks designed to existing regulatory standards when those casks are subjected to severe transportation accident conditions. Results of the initial study are expected to be submitted for an independent peer review before conclusions are reached. The final results will be used as a basis for correcting deficiencies and to demonstrate the degree of protection provided for shipments of spent nuclear fuel.

Materials Safety

In April 1985, the NRC proposed new regulations for well logging using radioactive sealed sources and tracers. Well logging is the measurement of the characteristics of the rock strata around a well by means of various instruments which are lowered down into the well. Approximately 100 public comments were received on the proposed regulations by the end of the comment period. A final rule should be published in 1986.

THERMAL-HYDRAULIC TRANSIENTS

Best-estimate systems codes and evaluation model computer codes are two of the basic computer tools used in analyzing nuclear power plant safety. Best-estimate systems codes offer a way to apply the results from reactor safety research to evaluations of accidents because they encompass the entire reactor coolant system. Evaluation model codes provide conservative analyses for use in independent audits of licensing calculations.

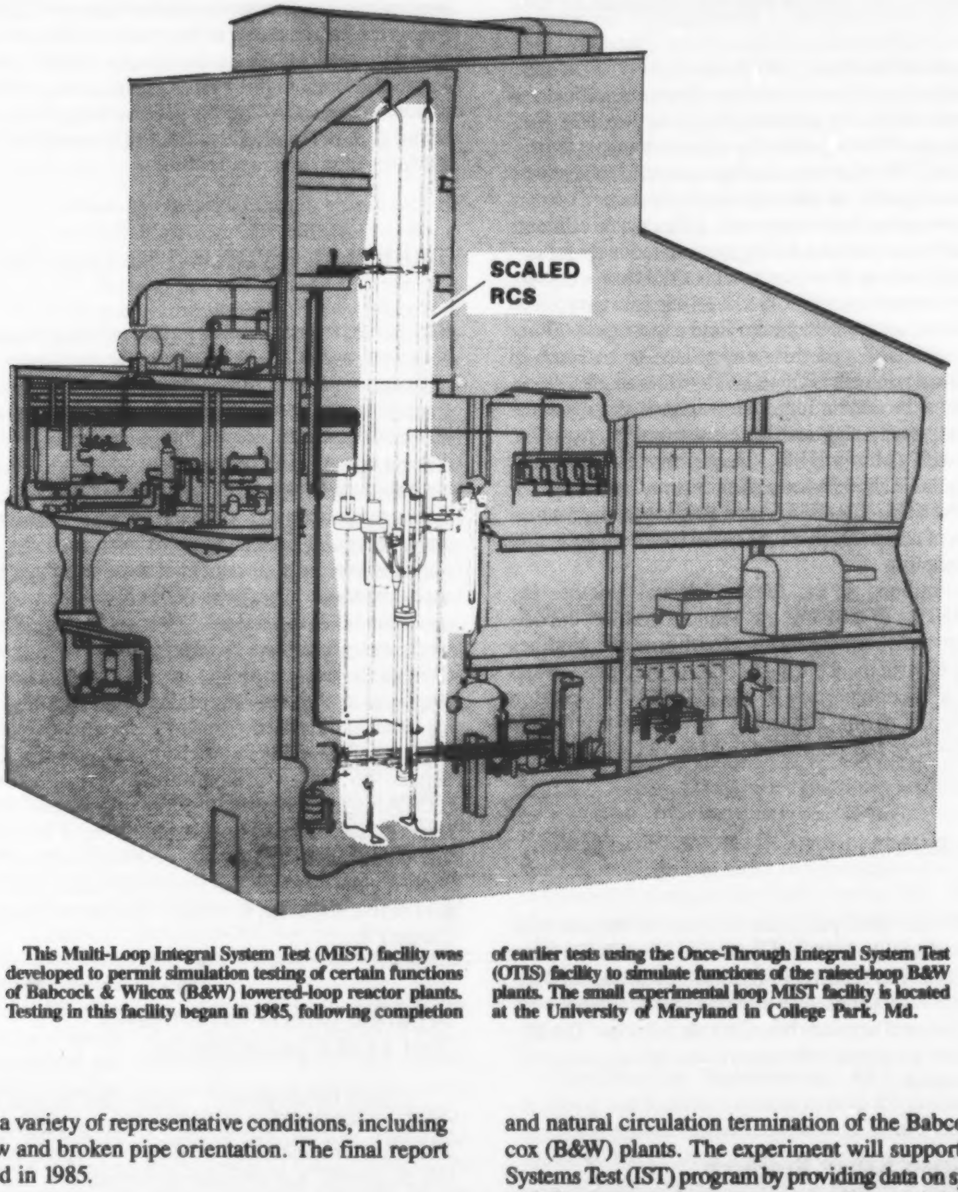
Reactor coolant experimental programs comprise the separate effects and integral systems tests needed to support the improvement and assessment of these computer codes. These experiments and computer codes assist the licensing staff in resolving licensing and safety issues. During 1985, additional work was performed to improve the usability of the codes through the nuclear plant analyzer and data bank programs. Application of the codes continues in support of such licensing issues as the effectiveness of upper plenum injection in Westinghouse 2-loop PWRs; revision to Appendix K; core liquid depression during a small-break loss-of-coolant accident (LOCA); and the June 9, 1985 loss of feedwater transient at the Davis-Besse plant (Ohio). Analytic input to the pressurized thermal shock studies was successfully completed in 1985. The nuclear plant analyzer was used to simulate a real reactor accident in training drills at the new NRC Operations Center (see Chapter 8).

Separate Effects Experiments And Model Development

Model Development. Most NRC model development takes place at universities and is aimed at supplementing separate effects experiments, helping to interpret data from larger test programs, and developing correlations based on a new understanding of the phenomenology (see the *1981 NRC Annual Report*, p. 124). During this past year, model development tasks were completed at Lehigh University, Pennsylvania State University, Northwestern University, the Massachusetts Institute of Technology, the University of California at Berkeley, and Purdue University.

THL/Critical Flow Experiments. Experiments were completed in 1984 in the Thermal-Hydraulic Loop (THL) at INEL, jointly funded by NRC and EPRI, to provide critical

Multi-Loop Integral System Test (MIST)



This Multi-Loop Integral System Test (MIST) facility was developed to permit simulation testing of certain functions of Babcock & Wilcox (B&W) lowered-loop reactor plants. Testing in this facility began in 1985, following completion of earlier tests using the Once-Through Integral System Test (OTIS) facility to simulate functions of the raised-loop B&W plants. The small experimental loop MIST facility is located at the University of Maryland in College Park, Md.

flow data for a variety of representative conditions, including stratified flow and broken pipe orientation. The final report was published in 1985.

Hot Leg U-Bend and Inverted Annular Flow Experiments. The hot leg U-bend and the inverted annular flow experiments being conducted at the Argonne National Laboratory are intended to assess scaling compromises in experimental facilities and to develop two-phase flow models and correlations to support LWR safety analyses. The hot leg U-bend experiment is conducted in an air-water experimental loop to study two-phase flow regimes, flow separation mechanisms and natural circulation termination of the Babcock and Wilcox (B&W) plants. The experiment will support the Integral Systems Test (IST) program by providing data on specific issues and phenomena relevant to post-small-break LOCA transients for B&W reactors. Parts were procured to construct the scaled hot leg U-bend as well as to check fluid property dependency of flow regime transition. The other experiment being conducted is the inverted annular flow experiment using Freon to study the post-critical heat flux flow regimes. The inlet to the test section is designed to give two-phase (slug, churn and bubbly) flow conditions resulting from steam generator tube rupture and steam line break. All testing is now finished, and final analyses of the data will be completed in 1986.

MB-2. A steam generator test program, Model Boiler-2 (MB-2), operated jointly by Westinghouse, EPRI and NRC, completed production of data simulating accident conditions resulting from steam generator tube rupture and steam line break. All testing is now finished, and final analyses of the data will be completed in 1986.

Integral Systems Experiments

The NRC has been the major source of support for the Loss-of-Fluid Test (LOFT) and Semiscale PWR test facilities at Idaho National Engineering Laboratory (INEL), although approximately 10 percent of LOFT support has come from foreign countries. Since early 1983, the LOFT facility has been operated by the U.S. Department of Energy for a consortium of which NRC is a member. The final LOFT test was performed in 1985, and the facility is now being decommissioned. Other United States integral facilities include the Full Integral Simulation Test (FIST), a BWR test facility which was supported almost equally by the NRC, EPRI and the General Electric Co. (GE); and the IST program, sponsored by Babcock & Wilcox (B&W) plant owners, B&W, EPRI and NRC. In addition, the NRC participates through international agreements in the 2D/3D facilities in West Germany and Japan, the ROSA-IV facility in Japan, and several other smaller facilities in Europe.

Semiscale. During 1985, two test series were performed in the Semiscale MOD-2C system. The first test series simulated steam line and feedwater line breaks coupled with appropriate compounding failures, e.g., loss of off-site power. Steam line breaks have the potential for overcooling the reactor coolant system at high pressures. This in turn could result in pressurized thermal shock to the reactor vessel. The first two tests in the series modeled offset shear of the steam line upstream and downstream of the flow control valve. Feedwater line breaks simulating 100, 50 and 10 percent of the feedwater line area were performed. Serious degradation of the steam generator heat transfer prior to core scram could result in reactor system overpressurization. Some primary system pressurization was observed during each feedwater line break test.

The principal value of the Semiscale MOD-2C design for these tests is to provide extensive measurements in the steam generators, as an aid to assessing the capability of NRC's best-estimate code and to improving staff understanding of the transient response. The effectiveness of a variety of recovery procedures based on operating reactor abnormal transient operational guidelines (ATOGs) was evaluated during the final stages of four of these experiments.

Two small-break LOCA tests were also performed during 1985 to assess core level depression and heatup behavior. These phenomena were first encountered during Semiscale test S-UT-8 performed in 1981. They have been of concern because NRC's best-estimate codes did not predict this response. The two tests performed in 1985 were both 5 percent cold leg breaks, with a 0.9 percent and 3.0 percent downcomer-to-upper head bypass flow, respectively. Two core level depressions and heatups were observed in the first test. The first heatup was caused by upper plenum pressurization—perhaps from manometric effects of liquid holdup in the loop prior to loop seal clearing—and the second was caused by net system coolant inventory depletion. The second test exhibited only the latter level depression and heatup. The RELAP5 computer code predicted no core heatup for either test. These tests serve as counterpart tests to those being performed in the Japanese ROSA-IV facility under a cooperative agreement with NRC.

IST Program. The Integral Systems Test (IST) program was initiated in 1983 to conduct tests in integral facilities representative of B&W plants. The program includes the Once-Through Integral Systems (OTIS) test facility, which simulates the raised-loop B&W plants, and the Multiloop Integral Systems Test (MIST) facility, which represents the lowered-loop B&W plants. During 1984, all planned OTIS tests were successfully conducted, and resultant data were used to verify advanced system codes. The effects of break size, break location, leak isolation, steam generator characteristics, feed and bleed cooling, and natural circulation cooldown were investigated by the 15 OTIS tests. The OTIS final analysis report was to be published at the end of 1985. Following the completion of the OTIS testing phase, the facility was modified to the 2x4 MIST facility configuration. Construction of the MIST facility was completed in September 1985. Shakedown testing started at the beginning of October 1985, and transient testing was scheduled for April 1986.

University of Maryland 2x4 B&W Simulation Loop. The University of Maryland (UM) 2x4 B&W simulation loop is an IST support facility that will provide experimental data to complement the MIST data base by addressing the effects of facility scale distortions and MIST atypicalities. The UM 2x4 B&W simulation loop is constructed to simulate the natural circulation and small-break LOCA behavior of a prototype lowered-loop B&W plant. All components are volume scaled in the ratio of 1:500. The loop has a maximum pressure of 300 psia. Construction of the facility was completed in June 1985. Shakedown testing was in progress at the end of the fiscal year, with transient testing scheduled for the beginning of fiscal year 1986.

2D/3D Program. Under this joint research program with Japan and Germany to study the refill and reflood phases of PWR LOCAs, the Japan Atomic Energy Research Institute (JAERI) completed the Core II test series in both the Slab Core Test Facility (SCTF) and the Cylindrical Core Test Facility (CCTF) and went on to construct the SCTF Core III facility, which was to be completed by the end of 1985. The primary objective of the SCTF Core III test series is to provide the heated core characteristic to the German Upper Plenum Test Facility (UPTF—see figure) so that the proper amount of water and steam can be injected into the UPTF vessel, simulating the reactor core. Of particular interest to the United States were the data JAERI obtained from five tests in the CCTF II test series, which employed the emergency core coolant (ECC) injection into the upper plenum. These data are useful to licensees for upper plenum injection plants in improving their emergency core cooling system evaluation models.

The UPTF is undergoing a series of shakedown and acceptance tests and will be ready for main tests in early to mid-1986.

The Upper Plenum Test Facility at Mannheim, Federal Republic of Germany, is a four-loop, full-scale facility with a 1,300-megawatt capacity. It features a simulated core and steam generators.

The UPTF is a full-scale test facility that will provide, among other things, data on de-entrainment of liquid droplets in the upper plenum, ECC bypass around the downcomer, and the countercurrent flow limitation in hot legs.

Continuing Experimental Capability. By the end of 1987, all major integral thermal-hydraulic test facilities in the United States are scheduled to be shut down. This move will affect the NRC's ability for timely resolution of future unforeseen safety issues with a high level of confidence The NRC is therefore evaluating available options for maintaining a testing capability for each of the major light-water reactor (LWR) design types. A scaling study was initiated in 1985 to evaluate capabilities and costs of four alternative scaling approaches to test facility design. This study will form one basis for a decision on future experimental facility needs and design approaches.

Code Assessment and Application

Code Improvement. Work continued on several best-estimate codes during 1985. Further improvements were made to TRAC-PFl/MODl, used to analyze system transients that require a complete simulation of PWR plant controls and balance-of-plant systems. This code is also capable of analyzing LOCAs, since it contains models similar to its predecessor codes, i.e., TRAC-PD2 and TRAC-PFl. The TRAC-BDl/MODl code, used to analyze the same aspects of BWRs, was completed during the report period, and development of an interim version of the fast-running TRAC-BFl was also completed. The COBRA-FS code to analyze flow blockage and rod swelling effects upon the cooling of a fuel assembly was also completed. In 1985, greater emphasis was placed upon making these codes easier to use and assisting code-users through guidelines, documents, newsletters and "trouble-shooting" resources. International code-users avail themselves of such services through the bilateral technology exchange agreements discussed below. Domestic, nongovernment users can obtain such services, on a cost reimbursable basis, through participation in domestic code-users groups.

Code Assessment. Several efforts are under way to develop a methodology to quantify the accuracy of thermal-hydraulic transient codes. An accepted method will be adopted in 1986 and will be applied to both international and domestic code assessment calculations.

Code Applications. These computer codes continued to be used to address licensing concerns. The TRAC-PFl/MODl computer code was used to address licensing issues raised as a result of the June 9, 1985 loss-of-feedwater incident at the Davis-Besse plant (Ohio). The TRAC-PFl/MODl code is also being used for calculations of experimental systems and large PWRs to support experimental programs such as MIST, OTIS, 2D/3D and ROSA-IV.

International Thermal-Hydraulic Agreements. Results of the code assessment work from several cooperative bilateral thermal-hydraulic research agreements will be made available worldwide, thereby enhancing the safety of nuclear reactors operating in all countries.

The first meeting of the International Code Assessment Program (ICAP) was held in April 1985. The NRC presented a draft document on procedures for execution of ICAP. This document was discussed with participants and is being revised based on the comments received. The procedures will be used to integrate the results of the code assessment work to be performed by 10-15 different countries to maximize the benefits to all participants.

Pressurized Thermal Shock. In June of 1985, the Comnission approved the pressurized thermal shock (PTS) rule. 'his rule sets limits on reactor vessel brittleness and requires icensees whose plants will exceed these limits during their perating lifetime to perform detailed analyses to determine he likelihood that cracking could occur in the reactor presure vessel by inadvertent overcooling. The regulatory guiance that recommends how these analyses are to be performed s modeled after pilot studies performed for three specific uclear reactor systems. The detailed thermal-hydraulic calulations for these pilot studies were performed using the 'RAC and RELAP5 computer codes. The design information or these pilot studies was provided by Duke Power Company for Babcock and Wilcox plants), Baltimore Gas and Electric Company (for Combustion Engineering plants), and Carolina 'ower and Light Company (for Westinghouse plants). These alculations were supported by thermal-fluid-mixing experinents performed at Purdue University and Creare Research, nc. The international interest generated by these pilot studies as resulted in the exchange of related reactor safety research nformation among the United States, West Germany, and 'inland.

Plant Analyzer and Data Bank. The plant analyzer ncludes calculational tools by which to easily and accurately nalyze plant transients and make swift regulatory decisions n the safety of operations at that plant, as well as others of imilar design.

There are four purposes served by the nuclear plant analyzer NPA):

(1) To reduce the man-hours required to prepare computer input.

(2) To provide on-line interactive computer capability to simulate reactor operator actions.

(3) To speed up existing TRAC and RELAP5 computer programs to provide a technical basis in a timely manner for licensing decisions and to cut costs.

(4) To provide a color graphics computer output picture of the plant, showing at a glance the instantaneous thermodynamic state of the fluid and vapor throughout the primary or secondary loop and thus to speed up analysis of results.

During 1985, stand-alone color-graphic replay capabilities t the work station were developed to save mainframe computer nd communications costs. The color schematic display for a eactor can now be defined interactively, saving user time. 'hree-dimensional cutaway schematics were developed to porray results of three-dimensional calculations. Computer run mes have been reduced by a factor of three compared to RAC-PF1, by changing the numerical technique. Conversion a parallel computer was demonstrated and, on completion 1986, will achieve a speedup factor of 10.

In 1985, the NPA was used by the NRR staff to analyze the avis-Besse loss-of-feedwater event and to evaluate operator uidelines for main steam line break and steam generator tube upture events. The NPA was also used to simulate a reactor in transient for two emergency preparedness exercises at the NRC Operations Center. At INEL, the NPA graphics system was used to evaluate Three Mile Island core data. Similarly, the NPA was used with RELAP5 to compare predicted and actual results at the INEL Semiscale test facility.

SEVERE ACCIDENTS

Accident Likelihood Evaluation

In 1985, the accident likelihood evaluation program continued to provide information on LWR accident sequences. In particular, the Accident Sequence Evaluation Program (ASEP) has been generating data relevant to the source term reassessment, the NRC/IDCOR interaction regarding severe accident issues, preparation of the NRC risk reference document, the proposed severe accident policy statement, and the formulation of a computerized PRA information base on a microcomputer. Because of the need for more current and plant-specific accident risk information, the development of an NRC risk reference document was begun in 1985. More detailed accident likelihood analyses will be prepared for the six reference plants selected for this study (Surry (Va.), Peach Bottom (Pa.), Sequoyah (Tenn.), Grand Gulf (Miss.), Zion (Ill.) and LaSalle (Ill.)). Among other things, this document will provide an independent NRC audit capability for evaluating industry proposals for resolving severe accident issues. Work completed to date includes reference plant visits to obtain current plant-specific procedures and designs. Modeling of plant safety systems and quantification of accident sequence likelihood were also initiated. As part of the program, a final report was published summarizing the dominant accident sequence information from 12 PRAs and identifying the factors that constitute the sequence likelihood. Methods for accident sequence precursor analysis were completed and transferred to NRC's Office for Analysis and Evaluation of Operational Data for implementation.

Severe Accident Sequence Analysis Program

Under the Severe Accident Sequence Analysis (SASA) research program, assessments of power reactor response to various possible sequences of events beyond the design basis accidents are continuing. Studies have been completed at five of the national laboratories—Los Alamos, Idaho, Oak Ridge, Sandia and Brookhaven.

The following is a summary of the research performed by each laboratory.

The Los Alamos program for 1985:

- The final report on dominant accident sequences in Oconee 1 (S.C.) pressurized water reactor (NUREG/CR-4140) was issued.

- Analysis of potential core damage sequence during station blackout for Oconee 1 using the TRAC/MELPROG code is near completion. A summary of TRAC/MELPROG analyses of station blackout transients in Oconee 1 was submitted to the Thirteenth Water Reactor Safety Research Information Meeting in October 1985.

The Brookhaven National Laboratory program in 1985:

- Best-estimate calculations for the Browns Ferry Unit 1 (Ala.) anticipated transients without scram (ATWS) with the RAMONA-3B code were completed. Browns Ferry Cycle 5 nuclear data or cross sections used in the RAMONA-3B calculations for ATWS transients were generated at Brookhaven.
- A draft report on RAMONA-3B code calculations for Browns Ferry ATWS is near completion.
- RAMONA-3B calculations have been performed to study the effect of manual control rod insertion during the ATWS involving main steam isolation valve closure with both level and pressure control.

The INEL (Idaho) program in 1985:

- The integrated SCDAP/RELAP5/TRAPMELT code described below is being used to analyze core damage sequences in PWRs. An advantage of using the integrated code is that it permits a mechanistic approach and provides feedback effects for the analysis of thermal-hydraulics, core damage and fission products. Detailed station blackout analyses on the Bellefonte PWR (Ala.) using the integrated code are proceeding.
- Four sequences initiated by feedwater transients were analyzed to support the Bellefonte nuclear plant PRA being developed at TVA using the RELAP5/MOD2 code. The methods used to model and calculate these sequences have been discussed and agreed upon with TVA personnel.
- Analyses of the Browns Ferry Unit 1 BWR (Ala.) ATWS using the RELAP5 and SCDAP codes to determine reactor system and core response were completed. A draft report on SASA program ATWS simulations for Browns Ferry Nuclear Plant Unit 1 was completed.

The Sandia program for 1985:

- The linking of core melt-concrete interactions with MARCH for application in risk studies was completed. There was a major review and analysis of large, dry containments to identify potential limited local regions with detonable concentrations of hydrogen during severe accident sequences. Negligible risk has been found to be present, except for possible hydrogen stratification. An evaluation of the kinetics of diffusion and mixing is near completion.
- Studies were begun on station blackout with loss of turbine-driven auxiliary feedwater and on the small-break LOCA with failure of ECC injection for Bellefonte Unit 1.

The ORNL program for 1985:

- Analyses of dominant severe accident sequences for the Browns Ferry Unit 1 plant continued. The Station Blackout Study (1981) is being repeated to assess the differences attributable to the improved models that have been developed.
- The first assessment of a common mode three-plant failure (loss of instrument air) was completed.
- Melt/concrete interaction effects (CORCON) have been linked to MARCH.
- The high-efficiency particulate air (HEPA) filter plugging tests were completed and showed adequate resistance to tearing and collapse.
- The SASA compendium of BWR results was applied to the ASEP reevaluation of sequences and to the formulation of the rebaselining of the PWR plants for risk.

Behavior of Damaged Fuel

Severe Fuel Damage Test. The last Severe Fuel Damage Test (SFD1-4) was successfully performed in the Power Burst Facility (PBF) at INEL in February 1985. The test fuel bundle and experiment procedures were very similar to the previous experiment (see the 1984 NRC Annual Report, p. 151), except that the bundle contained four silver-indium-cadmium control rod tubes. Analyses of measurements indicate that the control rods melted and extensive zircaloy oxidation and fuel damage occurred as planned.

ACRR Experiment on Debris Formation. The debris formation (DF-2) experiment was successfully performed in the Annular Core Research Reactor (ACRR) at Sandia in October 1984. DF-2 was the second in a short series of separate effects experiments (see the 1984 NRC Annual Report, p. 151). It examined the effects of higher initial cladding pre-oxidation (15 percent vs. 10 percent) on the severity of the fuel damage processes. Post-experiment radiographs show less fuel damage for DF-2 than for DF-1.

Coolant Boil-Away and Damage Progression Test. The Full-Length High-Temperature Test 1 (FLHT-1) was successfully performed in the National Research Universal (NRU) reactor in Canada by PNL in March 1985. A peak temperature of 2,100 Kelvin was reached. The next test (FLHT-2) is expected to reach a peak temperature of 2,400 to 2,500 Kelvin. These tests are reducing uncertainties associated with length and power distribution scaling factors, and they are enabling the interpretation of the results from small-scale separate effects experiments.

Severe Accident Analysis Code Development Program. An integrated SCDAP/ RELAP5 code was developed to model severe accident progression in the entire reactor coolant system of a PWR. The code predicts the release rates of radioactive materials and hydrogen gas from a damaged reactor core through a pipe break or a relief valve to the containment. It was used in July 1985 to make a pre-test prediction for the LOFT FP-2 core damage test; comparison with preliminary LOFT data was good.

Hydrogen Generation and Control

This program assesses both the consequences and methods used to control or mitigate deflagrations, diffusion flames and detonations that might be caused by hydrogen burns in LWR plants. HECTR Version 1.0 user's manual (NUREG/CR-3913) was issued. The HECTR computer code was developed at Sandia National Laboratories and is used in the analysis of nuclear reactor accidents involving the transport and combustion of hydrogen. The assessment of HECTR is ongoing, and it includes extensive use of the data from EPRI/NRC large-scale hydrogen combustion experiments performed at the Nevada Test Site (NTS) (NUREG/CR-4138). Using the HMS-Burn Code (NUREG/CR-4020) developed at Los Alamos, the transport and mixing models in HECTR were assessed and judged adequate for most containment analyses. HECTR was used to predict pressure-temperature response of an ice-condenser containment for 53 accident scenarios (NUREG/CR-3913).

The thermal response and survival of typical safety-related electrical equipment was tested in the Sandia Central Receiver Test Facility (CRTF) at radiant heat flux levels simulating the thermal heat flux expected for a hydrogen burn in reactor containment. A pressure transmitter, solenoid valve and electric cables were exposed to a thermal heat flux 300 percent of that predicted for a 13-volume percent hydrogen deflagration in a PWR large, dry containment without experiencing major damage or functional impairment. The CRTF radiant heat flux was gauged by simulating the thermal heat flux measured in the NTS test series for a 13-volume percent pre-mixed hydrogen deflagration. The CRTF and NTS test series produced almost identical cable degradation where the jacket split and ignited but the electrical insulation was undamaged. The results of both test series are being evaluated in an attempt to determine if critical electrical equipment would survive a hydrotion in the event of a 75 percent core metal-water reaction in a PWR large, dry, full-scale containment. This information will be considered along with other research information in determining whether or not changes are required to the regulations covering hydrogen control (50.44 of 10 CFR Part 50) as they apply to PWRs with large, dry containments.

Experiments in the steam/hydrogen flame jet facility to investigate the properties of diffusion flames were documented (NUREG/CR-3638). These data were used as a basis for the preliminary diffusion flame model in HECTR. The HECTR diffusion flame model gives the NRC the capability to assess the threat to safety equipment and containment penetrations caused by thermal loading from standing diffusion flames. Information was obtained on the feasibility of deliberate ignition schemes that will function during station blackout through the use of platinum catalytic igniters. Information was also obtained on the flow of air in nuclear reactor containment buildings as a result of the introduction of water sprays and was coordinated with licensing activities. Experiments were conducted to investigate detonability of hydrogen-air-steam mixtures to resolve the issue of hydrogen control for PWR large, dry containments for both equipment survival and local detonation. Flame acceleration experiments were performed to resolve the outstanding issue of flame acceleration in PWR ice-condenser containments. Draft reports have been prepared to close out these licensing issues.

Fuel-Structure Interaction

Results of molten fuel/concrete interaction tests and molten steel/concrete interaction tests with delayed addition of water were analyzed and documented. These tests showed that molten fuel attacks concrete at a much slower rate than molten steel does. The addition of water did not appreciably slow down the rate of the concrete ablation. A cooperative agreement has been negotiated with the Federal Republic of Germany to conduct two steel/concrete tests at their BETA facility, using concrete crucibles constructed of materials typical of United States nuclear plants. Preparation for molten fuel/concrete tests with simulated decay heat has been completed; the first test in the series was expected to take place early in 1986. Initial data on post-solidification behavior of the molten core materials (hot solid tests) are being evaluated. The tests will continue through 1986.

The interaction chamber for tests to quantify direct heating of the containment atmosphere by pressurized ejection of molten core materials has been procured. Sandia will begin such tests in the second quarter of 1986. Tests to investigate mechanisms for aerosol generation during a core/concrete interaction are being planned.

Containment Analysis

The CONTAIN computer code analyzes the complex physical, chemical and thermal phenomena that characterize the interactions associated with severe accident conditions inside a multi-compartmented reactor containment system. No in-vessel processes are considered. CONTAIN input is derived from other codes dealing with earlier phases of the accident. Pressure and thermal loading are computed, and evolution of the radiological source term is tracked. This information is made available for the computation of environmental consequences in the event of containment failure. Since its release in August 1984, 24 copies of CONTAIN have been distributed to Government, private and industrial laboratories throughout the United States and five foreign countries. During 1985, the efficiency of the code was improved by the design and implementation of an implicit solution method. CONTAIN became available for BWR analysis with the installation of a suppression pool model. User support expanded through the presentation of two workshops, which included the opportunity for interactive training experience. A program for the quantitative evaluation of containment loading analysis (QUECLA) was initiated.

The CORCON code is a computer simulation of a pool of molten (or partially molten) core debris in a concrete cavity. Initial and boundary conditions must be provided as input, e.g., cavity shape, chemical composition of the concrete and debris, as well as the mass and temperature of the melt. As thermal ablation of the concrete advances, the code calculates basemat penetration, the evolution of combustible and noncondensable gases and the transfer of radiant and convective heat from the pool surface to the upper cavity environment. CORCON has been coupled to the CONTAIN and MARCH codes to facilitate the calculation of containment loading under accident conditions. It serves as a driver for the VANESA (see below) model for fission product release and aerosol generation at the debris pool surface. CORCON MOD2 has been integrated into the NRC suite of codes used to predict the potential radiological source terms that may accompany severe reactor accidents. During 1985, the principal author of CORCON was assigned to the large-scale core/concrete interaction research project at the Kernforschungszentrum, Karlsruhe, Federal Republic of Germany. This intimate involvement in the BETA project contributed significantly to the refining and validating of the CORCON code. A cooperative code comparison exercise intended to provide developmental guidance and increased understanding of core/concrete interactions was initiated.

The VANESA code is a mathematical model that simulates the phenomena that occur when gas bubbles sparge through a pool of molten core debris (such as that represented by the CORCON code). Radioactive fission product vapors, as well as other components of the melt, equilibrate with the bubble gas and are carried to the surface and released to the atmosphere. Fragments of bubbles bursting at the surface become airborne and contribute to the aerosols resulting from vapor condensation. This information provides input for CONTAIN and other codes used to calculate the radiological source term in the containment environment. During 1985, an in-depth technical review of the fundamental physics and chemistry underlying the VANESA model was conducted. An exhaustive report documenting the technical basis, with code-user information, was prepared and released for peer review.

Fission Product Release and Transport

This program develops computer models and obtains experimental data to determine the radiological "source term," which is defined as the quantity, timing and characteristics of the release of radioactive material to the environment following a core melt accident. The research is used to develop reactor siting policy, emergency planning and response requirements, PRA consequence calculational methods, and equipment qualification standards.

Fission Product Experiments. A high-pressure fission product release measurement program was initiated at Battelle Columbus Laboratories to study the effect of pressure on the rate and chemical form of fission product release. The effect of pressure is not considered in current fission product release computer codes. Calculations based on thermodynamic analysis show the significant impact of pressure on the prediction of the release of barium and strontium. The program results will be relevant to severe accident sequences where high pressure prevails in the reactor vessel leading to vessel failure. One of these sequences is the loss of all electric power combined with loss of auxiliary feedwater to the steam generators (station blackout sequence).

Source Term Reassessment. A major milestone in the severe accident research program was reached with the issuance for comment of draft NUREG-0956 in July 1985. NUREG-0956 describes NRC staff and contractor efforts to reassess and update the agency's analytical procedures for estimating accident source terms for nuclear power plants. The effort included development of a new source term analytical procedure—a set of computer codes—that is intended (1) to replace the methods of the Reactor Safety Study (WASH-1400) and (2) to be used in reassessing the use of TID-14844 assumptions (10 CFR Part 100). Both the Reactor Safety Study methods and the TID-14844 assumptions are currently in use in many areas of regulatory practice. Improved analytical procedures are needed in some areas of regulation to resolve safety questions, to assess the adequacy of current regulatory practices, and to implement the Commission's severe accident policy statement. NUREG-0956 describes the development of these codes, the calculation of source terms for specific cases, the peer review of this work, some perspectives on the overall impact of new source terms on plant risk, the plans for related research projects, and the conclusions and recommendations resulting from the effort. The new analytical procedures described in NUREG-0956 are currently being used by the NRC in a major risk rebaselining effort involving six reference plants. Detailed plans are also being formulated by NRR for further use of new source term information in a number of regulatory areas.

Aerosol Experiments. The NRC is participating in an internationally sponsored project called the LWR aerosol containment experiments, being conducted in Richland, Wash., by the Westinghouse Hanford Company. The six experiments are planned to investigate inherent aerosol retention behavior in the containment or auxiliary buildings for postulated high-consequence accident conditions, when the existing data base is inadequate, and also to provide a data base for validating containment aerosol and related thermal-hydraulic computer codes.

ORNL is conducting experiments in the aerosol moisture interaction test vessel on the effects of relative humidity, aerosol composition, and aerosol concentration on aerosol characteristics and behavior. The first three tests have been completed. ORNL is also conducting mixed-component aerosol experiments in the Nuclear Safety Pilot Plant containment vessel using a steam atmosphere.

Containment Failure Mode

Activity has continued on a set of programs whose objectives are to provide the data base required for the qualification of methods for predicting the response of LWR containment buildings during severe accidents (those beyond design basis events) and extreme earthquakes. This set of programs is examining the modes of containment failure that would result in the release of radioactive materials beyond the containment boundary. These modes include structural failure of the containment building, leakage through or past the penetrations (electrical or mechanical), failure of containment isolation systems, or failure of the basemat by the molten reactor core.

Most of the effort in 1985 centered on the tests-to-failure of large models of containment structures. A 1:8-scale steel containment model (see figure) was tested to determine its response to pressure levels exceeding the design basis. Extensive structural analyses of the model were performed prior to the test. A number of penetrations were present in this experimental model, including operable equipment hatches with single '0' ring seals, personnel lock representation and a constrained pipe. The model was built to ASME Code specifications with a design pressure of 40 psig. An extensive structural data base was generated during the high-pressure test of the 1:8-scale steel containment model conducted November 15-17, 1984, at Sandia. Data was recorded at 21 different pressure levels up to and including 190 psig, which is 4.75 times the design pressure. Strains of nearly 6 percent and displacements exceeding two inches were measured. The model ruptured after the pressure in the model was increased to 195 psig. No significant leakage was detected up to this point, although the measured displacements around the equipment hatch indicated that leakage was imminent.

A conceptual design for a 1/6-scale model typical of reinforced concrete containments in the United States has been completed. A subcontract for the final design and construction of the model has been placed with United Engineers and Constructors (UEC). The contract with UEC also includes a series of pre-construction tests to ensure that a satisfactory model can be constructed and that the behavior of certain key areas, e.g., liner-stud-concrete anchorage, are representative of actual containments.

In 1985, a D.G. O'Brien electrical penetration assembly, identical to those used in many nuclear plants, was tested at Sandia for leakage and functional electrical behavior when exposed over a 10-day period to a simulated PWR severe accident environment of 360F and 150 psia. These environmental parameters comprise the conditions calculated to develop inside containment for the dominant PWR severe accident scenarios involving core melt and vessel failure. A negligible increase in leakage was observed, indicating that containment integrity would have been maintained. Tests of other currently used electrical penetration assembly designs under BWR severe accident conditions are planned for 1986.

Fission Product Control

Most engineered-safety-feature (ESF) systems are likely to be operational even during postulated accidents substantially more severe than current design basis accidents. However, there may be a substantial variation in the effectiveness of fission product removal of various ESF systems under conditions exceeding their design basis. A program is in progress to facilitate review and evaluation of ESF-system behavior under severe accident conditions. Two reports covering (1) technical bases and user's manual for the prototype of a suppression pool aerosol removal code (SPARC) (NUREG/CR-3317) and (2) ICEDF, a code for aerosol particle capture in ice compartments (NUREG/CR-4130) were published during the report period.

Accident Consequences and Risk Reevaluation

Risks of accidents affecting public health involve both the probabilities of radioactive releases caused by reactor accidents (the so-called "source terms") and the probabilities of various off-site consequences associated with these source terms. The magnitude of those consequences depends on such factors as the weather, atmospheric transport conditions, distance from the reactor, and emergency response by the public.

An improved computer code for estimating the consequences of severe reactor accidents (MACCS) was completed during 1985. This code, which replaced the CRAC code originally developed during the 1974 Reactor Safety Study, is being used

This large steel containment model, about 1/8th the size of the typical U.S. hybrid steel containments, was successfully tested by internal pressurization with nitrogen gas, at the Sandia National Laboratories, Albuquerque, N.M. The model was designed and fabricated by the Chicago Bridge and Iron Company, with a design pressure of 40 psig. It is about 14 feet in diameter and 30 feet high.

to provide improved estimates of the cost effectiveness of various accident prevention and mitigation features under consideration for application to operating reactors.

In addition, a major study of models used to predict the health effects of exposure to or ingestion of radioactive materials was completed during 1985, and public comments were requested. This study was commissioned by the NRC and was conducted by the Harvard University School of Public Health, in cooperation with a score of eminent health physics practitioners and medical experts in the field. The results indicate that radiological risks may be slightly higher than previously estimated, but the uncertainties are within the estimated overall range for risk estimates for the reactor industry. Applications of these codes and models are currently being made to such major subject areas as emergency planning, trial use of NRC's draft safety goals, risk estimate uncertainty analysis, staff environmental reports and staff hearing testimony.

Value-Impact Analysis

The NRC Office of Nuclear Regulatory Research has as one of its concerns the development and implementation of systematic methods that facilitate NRC decisionmaking. To date the program has provided insights into NRC perceptions and requirements for risk-related decisionmaking and conducted specific research tasks in support of decisionmaking processes. During the past year, the Commission has initiated and completed several safety-related regulatory analyses demonstrating the methods prescribed in the value-impact handbook (NUREG/CR-3568). The methods and procedures of the handbook have also been incorporated in the revised regulatory analysis guidelines (NUREG/BR-0058) for use by NRC staff and industry in evaluating the need for and effectiveness of a variety of regulatory actions; the latter would include rulemaking, standards development and backfitting safety improvements on nuclear operating plants. The value-impact handbook is playing a key role in supporting recent backfitting initiatives. Workshops were held in the Regions to demonstrate to the staff the use of existing cost-benefit methods and resources available to assist in performing high-quality regulatory analyses, which as a matter of policy must accompany proposed rules, guides and plant-specific backfits.

RADIATION PROTECTION AND HEALTH EFFECTS

Radiation Protection Standards

Revision of 10 CFR Part 20. The NRC staff completed preparation of a complete review of 10 CFR Part 20, "Standards for Protection Against Radiation" (see 1984 NRC Annual Report, p. 154). This revision reflects developments in radiation protection and advances in related sciences that have occurred since the issuance, nearly 30 years ago, of the present version of 10 CFR Part 20.

Decommissioning. During 1985, work continued on the development of residual radioactivity limits for decommissioning. The effort included working with the U.S. Environmental Protection Agency's (EPA's) program to develop Federal guidance on generally applicable environmental concentrations of residual radioactivity. Residual radioactive contamination limits are needed to ensure that the radioactivity levels associated with buildings, structures, equipment, materials, and land used in NRC-licensed activities following decommissioning are low enough to pose no undue risk to public health.

Petitions. The NRC has been petitioned to modify the present requirements for the calibration of teletherapy units covered under 10 CFR Part 35, "Human Uses of Byproduct Material." In 1985, a study to develop a less time-consuming and less expensive means to check teletherapy calibration was completed. A report of this study (NUREG/CR-4131) was published.

Two petitions for rulemaking involving 10 CFR Part 35 were processed during 1985. One petition requested the addition of iridium-192 wire to the list of sealed sources that could be implanted for treatment of cancer. The petitioner could not ensure that licensees could maintain accountability for small pieces cut from the wire or that contamination from the cut ends would be negligible; consequently, the petition was withdrawn. Another petition requested NRC to allow health professionals other thaen physicians to be licensed to use a bone mineral analyzer. Since physicians are licensed only by individual States to practice medicine in this manner, i.e., diagnose disease and initiate therapy, the petition was denied.

Radiation Protection Research

Metabolism and Internal Dosimetry. An interim report (NUREG/CR-4208) for the research project on gastrointestinal absorption of actinide was published in April 1985. A Research Information Letter (RIL #143) summarized the results of studies on the gastrointestinal absorption of plutonium in mice, rats and dogs. Several values of f1, the fraction transferred across the gut, were recommended for application to certain dose assessments. Additional studies are being conducted in baboons to provide for better interspecies extrapolation.

Environmental Pathways. NUREG/CR-3981, describing a research project on bioaccumulation of phosphorus-32 in fish, was published in February 1985. The results were discussed in RIL #141, which recommended use of a bioaccumulation factor 20 times lower than the stable phosphorus value used in Regulatory Guide 1.109. As a result of this study, surveillance requirements being considered for phosphorus-32 measurements in effluents from nuclear power plants were not instituted.

Other projects continuing through 1985 included medical evaluation and autopsies for workers occupationally exposed to thorium, metabolic studies of inhaled yellowcake in dogs, and metabolic studies of actinide and rare earth uptake and retention in monkeys.

Health Risk Assessments

Radon. Exposure to radon gas and radon progeny has been associated with an increased incidence of lung cancer. In order to improve health risk assessments in populations exposed to these carcinogens, a study of radium dial painters who were exposed to radiation from radon daughters in early adult life was continued. A paper on radon risk assessment was given at an International Conference on Occupational Safety in Mining and published in the conference proceedings.

Severe Accident Health Effects Model. Work continued on updating the health effects model that was used in the Reactor Safety Study (WASH-1400) to estimate the consequences of postulated severe accidents at nuclear reactors.

Other continuing projects included studies of the relative biological effectiveness of neutrons, using mice, and studies of the early effects of inhaled radionuclides (alone and combined with external irradiation), using rats and dogs.

Cooperative Efforts

In 1985, the NRC health effects program was closely coordinated with other Federal programs and with national and international scientific organizations concerned with radiation research and protection. Broader areas of mutual interest were coordinated through participation on the Committee on Interagency Radiation Research and Policy Coordination, which operates under the auspices of the Office of Science and Technology Policy. Radiation protection programs were coordinated with the National Council on Radiation Protection and Measurements and by participation on interagency working groups established by the EPA to develop Federal guidance on radiation protection matters.

Specific research areas were coordinated through meetings and joint programs with the Department of Defense (DOD) and Department of Energy (DOE), the EPA, the National Science Foundation, and the National Institutes of Health. An NRC- and EPA-funded study by the National Academy of Sciences to develop a report on the biological effects of internally deposited alpha-emitting radionuclides and their decay products is being continued. This study will be used in numerous NRC programs, such as the high-level-waste program, that require assessments of the genetic and carcinogenic effects of alpha radiation. Particular emphasis was also given to quantification of health risks of exposure to internal alpha emitters in the American Statistical Association Conference on Radiation and Health, which considered problems in the quantification of radiation health effects and was partially supported by NRC funding.

Occupational Radiation Protection

Health Physics Measurements Improvement. A study completed by Lawrence Livermore National Laboratory, NUREG/CR-4239, evaluated the ability of commonly used health physics survey instruments to determine dose rate at specific tissue depths. In addition, data on calculated health effects were used to evaluate the ability of these instruments to predict such effects. A task completed by the National Bureau of Standards (NBS), NUREG/CR-4266, concerns the development of a facility to calibrate beta sources and transfer instruments sent to NBS by commercial calibration services or by NRC licensees. These capabilities for national standards for beta radiation will improve the accuracy of radiation surveys that NRC licensees are required to perform.

Research was completed on ultrasensitive analysis procedures for improving detection capabilities for uranium, plutonium, and thorium by resonance ionization spectroscopy. The results of this study were published in NUREG/CR-4419.

Personnel Dosimetry. Rulemaking currently in progress would require NRC licensees to use the services of accredited personnel dosimetry processors or become accredited themselves to process the dosimeters that are provided to meet the requirements of 10 CFR Part 20. The accreditation program is currently operated by the NBS under the National Voluntary Laboratory Accreditation program. (See 1984 NRC Annual Report, p. 156.) There are currently 23 accredited processors and another 25 enrolled in the program. It is expected that approximately 40 additional processors will participate if the program becomes mandatory. The purpose of the program is to achieve and maintain an acceptably high level of performance among personnel dosimetry processors.

A study completed by the Pacific Northwest Laboratory (PNL), NUREG/CR-3609, indicated that the thermoluminescent albedo dosimeter is an appropriate personnel neutron dosimeter and that the most appropriate calibration source is the source normally employed in well-logging operations.

A program jointly funded by the NRC and DOE to assist in the development of a consensus performance standard for bioassay laboratories is now providing data for consideration by the organization developing the standard (Health Physics Society). Analysis of the first round of in vivo results from testing bioassay laboratories suggests that changes are needed for standardized definitions of acceptable detection levels.

NRC guidance is being developed for determining the skin dose from radioactive contaminant deposited on the skin. PNL has developed a new, improved method of calculating skin dose from fission and radioactive corrosion products. This method has been made available in a computer code, VARSKIN, that will compute the radiation dose at a specified depth in the skin from a radiation source that ranges in size from a point to a disk of 100-cm diameter. PNL has also recommended for NRC consideration skin contamination levels below which further decontamination should not be required.

A report prepared by Idaho National Engineering Laboratory, NUREG/CR-4033, presents recommendations on the application of personal air sampling devices in NRC licensee radiation protection programs. The performance tests show that available personal air samplers can provide a reliable, convenient method for breathing-zone sampling of workers in the work environment encountered in the licensed activities.

Occupational Exposure Data System. In 1969, the Atomic Energy Commission began requiring certain licensees to submit reports on occupational radiation doses received by workers. These data are collected and computerized in an NRC system called REIRS (Radiation Exposure Information Reporting System). The system provides a permanent record of the data and permits expeditious analyses of the two kinds of reports required (annual statistical summaries and individual termination reports).

Summaries of the annual statistical reports for 1983 revealed that the seven categories of licensees required to report monitored about 173,000 individuals, of whom about 60 percent received measurable doses. The workers received a collective dose of 61,000 person-rems, or an average dose of 0.6 rem-per-worker among those receiving a measurable dose (0.3 rem-per-monitored-person when the entire monitored population is considered). Eighty-seven percent of the persons monitored were in nuclear power reactors, and they incurred about 93 percent of the total annual collective dose. The average measurable dose received by individual nuclear power plant workers remained about 0.6 rem.

A second kind of exposure report required of certain NRC licensees provides identification and dose data each time that a monitored individual terminates employment with the licensee. Such information is now maintained for some 330,000 persons, most of whom were or are employed by nuclear power plants. The computerization of these data enables the NRC staff to respond quickly to requests for individual exposure histories and to analyze the data for trends. The data also help ensure that transient workers moving from plant to plant do not receive doses in excess of regulatory limits.

Respiratory Protection. Two reports were prepared to improve the protection to workers who use respiratory equipment in atmospheres containing radioactive materials. NUREG/CR-4111, prepared by Los Alamos National Laboratory, reports that in certain types of respirators a polydisperse aerosol can be safely substituted for the aerosol now required for such testing by 10 CFR Part 20. This will be a cost-effective improvement for many licensees, because equipment now used for quantitative-fit testing can easily be modified to perform this function. Another report prepared by Los Alamos, NUREG/CR-3953, provides workers and military personnel who must wear prescription eyeglasses with an acceptable method of sealing their respirators against their faces and temples.

Training. In cooperation with members of other NRC offices, two training reports were prepared. The first, NUREG-1127, provides general information and references useful for establishing or operating radiation safety training programs in plants that manufacture nuclear fuels or process uranium compounds that are used in the manufacture of nuclear fuels. The other report, NUREG-1134, was designed to facilitate planning and conducting radiation safety training in today's hospitals.

A research study on the possible use of robots to replace humans in reactor inspections in areas where high-level radiation might be found was undertaken in 1985. As a result of one robotics demonstration project, a tethered survey and inspection system (SURBOT, shown above) is scheduled for demonstration in an operating power plant in 1986. This system can be programmed to perform a variety of tasks without operator intervention.

Chemical Decontamination. The NRC continued to develop an information base for assessing the safety and effectiveness of decontamination alternatives for reducing occupational doses in nuclear power plants and for assessing the impact of decontamination on solidification systems. Observations and measurements were made during selected chemical decontamination activities at the Cooper (Neb.), Millstone Unit 2 (Conn.), and Quad Cities Unit 2 (Ill.) nuclear power plants. A report analyzing these results and similar earlier measurements conducted at other nuclear power stations will be published in 1986. A report published in 1985 described the impact of LWR decontamination on solidification, waste disposal and associated occupational exposure (NUREG/CR-3444).

Dose Reduction at Nuclear Power Plants. During 1985, Brookhaven National Laboratory (BNL) published several reports on occupational dose reduction at nuclear power plants (NUREG/CR-4254 and NUREG/CR-3469, Volume 2) and a comparative assessment of United States and foreign dose rates and dose reduction experience (NUREG/CR-4381). The BNL work identified numerous design concepts, operational techniques, and other dose reduction efforts that have been shown at individual plants to be successful and cost effective. BNL will continue looking at dose reduction research and development projects funded by industry and other agencies, as well as utility efforts, in order to identify additional cost-effective methods. The results of this surveillance project will be available to all nuclear power plant licensees and will provide the NRC staff with information to determine whether or not additional regulatory efforts are needed.

Robotics in Radiation Surveys and Reactor Inspection. In 1985, Phase I of the Small Business Innovative Research study on robotics in reactor inspection demonstrated the feasibility of using robots to replace workers for inspecting and monitoring radiation in areas of potentially high exposure (NUREG/CR-3717). The ongoing Phase II has comprised the design, construction, and demonstration testing of a tethered inspection robot (see figure) capable of taking measurements of radiation levels, sound, humidity, and temperature; performing air and contamination sampling; and allowing high resolution TV viewing of components in hazardous areas. The system is scheduled to be field tested in early 1986 at an operating nuclear power plant, and cost-benefit analyses will be performed at that time.

WASTE MANAGEMENT

NRC's waste management research seeks to develop and verify methods for predicting and assessing the performance of waste disposal facilities; it evaluates and confirms the data bases used in such performance assessment; it provides technical support to the licensing staff in their interactions with the DOE and the States (see Chapter 7); and it develops regulatory standards to support the licensing of facilities and methods for the disposal and management of high-level and low-level radioactive wastes.

High-Level Waste

The NRC has active research programs in hydrology, geology, materials science, geochemistry, and several other disciplines related to the management of high-level waste (HLW). The research combines theoretical study with laboratory and field experiments to identify the physical processes that control and determine repository performance in the types of geologic media found at sites currently under consideration by DOE. The ultimate goal of the NRC's waste management research is to provide the technical basis for the licensing staff to make its own independent judgment as to the appropriateness and adequacy of DOE's demonstration of compliance with 10 CFR Part 60 and the EPA's HLW standard.

Since transport by ground water is the most likely path by which radioactive nuclides from the disposed waste can reach the environment, the NRC is actively studying the movement of ground water in the types of media being considered by DOE. Experimental sites have been located in fractured rock, both above and below the water table, and field testing is being conducted to determine what type of measurements are needed to characterize the hydrology of fractured media and how measurement data should be analyzed to model ground water flow.

Investigating the performance that can be expected from the waste form and waste package is another major area of NRC's HLW research. NRC-sponsored research programs at national laboratories and other organizations are identifying and studying the mechanisms of waste package/waste form failure under expected repository conditions. These studies are essential if NRC is to be able to independently evaluate DOE's demonstration that the waste form and waste package comply with the containment and controlled release requirements of 10 CFR Part 60. During 1985, the corrosion research groups under contract to NRC used statistical methods of experimental design combined with cyclic voltammetry to assess the potential for failure of HLW metal overpacks by stress corrosion cracking and localized corrosion.

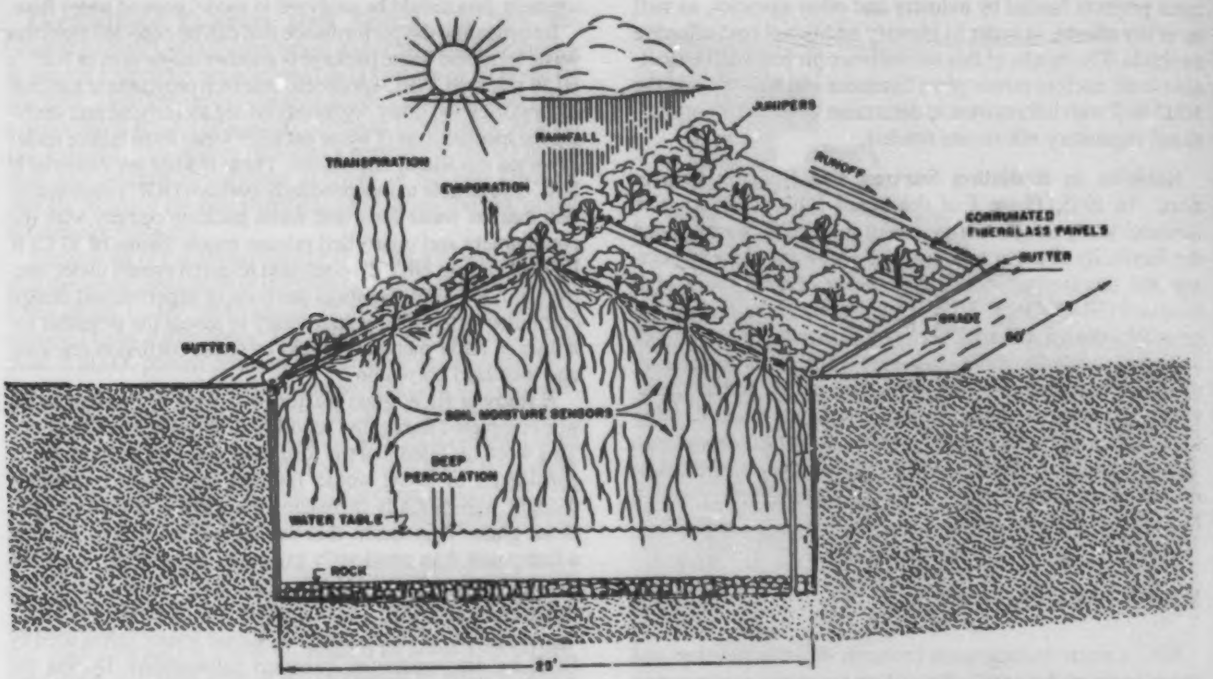
A study of HLW glass fracturing caused by thermal stress was completed during the report period (NUREG/CR-4198). This work revealed that the formulations of waste glass being considered by DOE would fracture more than previously thought, substantially increasing the surface-to-volume ratio of the glass. This means that waste glass is likely to leach at a faster rate than previously expected. In addition, leaching mechanisms for HLW glasses were modeled (NUREG/CR-3900). These models will be useful in assessing the validity of the radioactive release source terms used by DOE for environmental transport calculations. During the report year, the research on glass waste forms was phased out as attention was shifted to spent reactor fuel as the waste form.

The NRC has an active research program in the vital field of geochemistry related to the management of HLW. The program has led to an improved understanding of the movement of radionuclides through certain kinds of clay materials that may be used in repositories. In addition, a major error was identified and corrected in the data base used for prediction of rock/water/radionuclide reactions in a repository, and the effects of temperature on radionuclide solubility and mobility have been characterized.

Through NRC research, processes that cause short-term radionuclide mobility to differ from long-term natural mobility have been identified by experiment. A program of field tests of computer predictions of radionuclide transport is under way.

Research recently completed on the relationship between the chemistry of ground water and radionuclide mobility showed that the sulfate ion is relatively benign, but chemically reducing conditions—which were previously thought to be favorable for immobilizing radionuclides—may cause greatly increased radionuclide mobility in some cases. High radionuclide mobility has been observed in salt brines under slightly reducing conditions, which are the expected conditions for an HLW repository in salt.

NRC's research program on borehole sealing is providing experimental assessments of the performance of existing technology for sealing boreholes. The assessments will provide a factual data base for judging the acceptability of using presently available technology to reduce water flow (and radionuclide migration) through boreholes near an HLW repository. A major accomplishment of the borehole sealing research pro-



The NRC continued its studies and tests in 1985 to find ways to control water entry into low-level waste burial sites. In this demonstration of a bioengineered system, 90 percent of incoming precipitation is removed by direct runoff. The remaining 10 percent is removed from the system by plant transpiration plus evaporation ("evapotranspiration" or E.T.). On an annual basis, potential E.T. exceeds water available to plant roots, thus the initial water table will be depleted. Directed measurements of precipitation, runoff, soil moisture, and water table allow calculation of E.T. for complete measurement of water balance.

gram was the completion of a report (NUREG/CR-4174) on borehole seal performance. One significant finding was that cement plugs installed under water are likely to undergo channeling or piping along the interface between the plug and the host rock. Such channeling can cause drastic increases in hydraulic conductivity. Several alternative plug installation procedures intended to prevent channeling are being explored.

Experiments are also in progress to determine:

- The performance of boreholes sealed with commercially available bentonites.
- The effect of temperature on cement plug sealing performance.
- Size effects (length/diameter ratio) on the performance of cement plugs.
- A preliminary performance assessment of fracture grouting.
- The sealing performance of bentonite and crushed rock mixes.

The NRC is investigating the coupling among thermal, hydrological, mechanical, and chemical processes in a deep geologic repository. Lawrence Berkeley Laboratory is borrowing from experience with natural geothermal systems, field and laboratory experiments, and modeling studies they have conducted to identify and categorize which coupled processes are most important to repository performance.

In January 1985, the NRC published for public comment proposed procedural amendments to Part 60, dealing with site characterization and the participation of States and Indian tribes in the licensing process for an HLW repository. These amendments are needed to bring the procedures in Part 60 into conformity with those established by the Nuclear Waste Policy Act of 1982. The NRC received many comments from States and Indian tribes, all of which were duly considered. Final amendments have been sent to the Commission for action.

On the subject of the disposal of HLW in the unsaturated zone, final amendments to Part 60 were published in August 1985. These amendments will ensure that NRC regulations are applicable to all geologic repositories, whether sited in the saturated or unsaturated zone.

In March 1985, proposed Revision 1 to Regulatory Guide 4.17, which provides the standard format and content of a site characterization plan for HLW geologic repositories, was published for public comment. This revision provides updated guidance to DOE as it prepares site characterization plans.

Low-Level Waste

NRC research in support of licensing activities for low-level waste (LLW) disposal facilities is focused on (1) water entry into burial trenches, (2) performance of waste packages, (3) characterization of the LLW source term, (4) mechanisms for transport of radionuclides from the burial trenches, (5) the safety and performance of engineered enhancements and alternatives to conventional shallow land burial for LLW disposal, and (6) evaluation of the overall performance of disposal systems. This research will be useful not only to the NRC licensing staff but also to States facing similar regulatory efforts.

With regard to controlling water entry into burial trenches, the University of Arizona concluded field testing of LLW shallow land burial trench covers under both humid and arid conditions. Field testing demonstrated that covers of conventional compacted backfill soon failed as did covers designed to use soil arches and soil beams, constructed with the aid of geotextiles. The only successful cap was a soil cement beam. It exhibited no subsidence during a two-year period, and it inhibited water entry through the trench cover. The results of this project were published in NUREG/CR-4194.

In a related study, the University of California and the University of Maryland began field testing at Beltsville, Md., of a bio-engineered system to control water entry through trench covers (see figure). Preliminary results indicated that this combination of engineering and vegetation appears to be very resistant to failure from trench cover subsidence or deterioration of the material on the cover, because it effectively controls deep water percolation into the trenches. However, researchers at PNL, in looking at the role played by vegetation in enhancing radionuclide migration, found that plant roots exude mobile radionuclides to a degree greater than previously anticipated. The results of this research are being factored into geochemical/hydrologic transport models that will be used for predicting the performance of an LLW disposal site.

In May 1983, the NRC issued a technical position paper that specified minimum performance standards for LLW waste forms. Current waste forms in commercial use were tested by Brookhaven National Laboratory to ensure that leaching characteristics and compressive strength of the waste forms are consistent with the standards specified in the technical position. Various decontamination wastes from actual power plants using commercial solidification processes such as Lomi, Candecon, NS-1, and Citrox are being investigated.

There is great interest in either enhancements to shallow land burial or alternatives to shallow land burial as it is currently practiced. RES began a study in 1985 to identify and assess the importance of the key engineering design and safety features of a number of alternatives being considered by States and industry.

An NRC-sponsored cooperative project between Atomic Energy of Canada Ltd. (AECL) and PNL has been using data collected from 40 years of LLW waste disposal at AECL's Chalk River facility to assess techniques for modeling LLW site performance. PNL is approaching the problem as though dealing with a pristine site, prior to waste disposal. They will then progress through various stages of site analysis and site review, using those portions of the data base required to resolve indicated problems. They will use the full data base to assess the validity of the model predictions and analyses based on the smaller data sets. This project should provide important insight into the design of data evaluation programs for such sites and the reliability of predictions based on the data.

Uranium Recovery. A draft regulatory guide related to on-site meteorological measurement programs for uranium recovery facilities was issued for public comment in September 1985.

STANDARDS PROGRAMS

IAEA Reactor Safety Standards

The NRC continues to coordinate U.S. technical activities associated with the International Atomic Energy Agency's (IAEA) Nuclear Safety Standards program to develop safety codes of practice and safety guides for nuclear power plants. The codes and guides provide a basis for national regulation by developing countries of the design, construction and operation of these plants. In 1985, one safety guide was forwarded through the Senior Advisory Group and Technical Review Committee to the Director General of the IAEA. All the planned IAEA safety guides were undergoing review at year's end, with the NRC research staff coordinating the reviews within the U.S. A revision of one safety guide is under way in response to user request that more information on the commissioning phase be included. (See 1980 NRC Annual Report, p. 196.)

National Standards Program

The national standards program is conducted by the American National Standards Institute (ANSI). ANSI acts as a clearinghouse to coordinate the work of standards development in the private sector.

The NRC staff is active in the national standards program, particularly with respect to setting priorities so that regulatory views are known regarding the standards that can be most useful in protecting the public health and safety. NRC participation is based on the need for national standards to define acceptable ways of implementing the NRC's basic safety regulations.

Approximately 225 NRC staff members serve on working groups organized by technical and professional societies.

Proceedings and Litigation

CHAPTER 12

The first part of this chapter is a report on select proceedings involving the Atomic Safety and Licensing Board Panel and the Atomic Safety and Licensing Appeal Board. The second part is a judicial review of noteworthy litigation during the fiscal year involving the NRC, including cases pending and closed.

ATOMIC SAFETY AND LICENSING BOARD PANEL

Fiscal year 1985 was an unusually demanding one in the 23 year history of the Atomic Safety and Licensing Board Panel. Licensing Boards authorized operating licenses for 14 new nuclear power plants, completed a total of 23 complex proceedings, and authorized the restart of Three Mile Island Unit 1.

The Atomic Energy Act of 1954 requires that a public hearing be held on every application for a construction permit for a nuclear power plant or related facility. In certain circumstances, hearings are also held in connection with operating licenses, license amendments, antitrust issues, enforcement and civil penalty cases, and other matters as directed by the Commission. (See "The Licensing Process," in Chapter 2.) Boards composed of three administrative judges drawn from the Atomic Safety and Licensing Board Panel (ASLBP) perform the Commission's hearing function and render initial decisions in licensing cases; single administrative judges may also hear and decide other matters. These hearings are the Commission's principal public forum in which individuals and organizations can voice their interest in a particular licensing or enforcement issue and have their concerns adjudicated by an independent tribunal.

As of September 30, 1985, the panel included 22 permanent and 26 part-time administrative judges drawn from various professions. There were 18 lawyers, 15 environmental scientists, 7 engineers, 5 physicists, 1 medical doctor, 1 economist and 1 chemist. (See Appendix 2 for the names of panel members.) The Commission appoints administrative judges to the panel based upon recognized experience, achievement and independence in the appointee's field. Judges are assigned to cases in which their professional expertise will assist the board in resolving the issues to be litigated. Generally, boards consist of a lawyer as chairman, a nuclear engineer or reactor physicist, and an environmental scientist.

The hearing on a particular application for a nuclear facility license may be divided into several phases, each focusing on a particular licensing concern: health, safety, or the common defense and security aspects of the application, as required by the Atomic Energy Act; environmental considerations, as required by the National Environmental Policy Act (NEPA); and emergency planning requirements. These matters, as well as especially complex technical issues, are frequently the subject of separate initial decisions by the boards.

Administration

In March 1985, the panel sent the Commission the first five-year projection of workload and resource needs by a single office. The panel's Five Year Plan addressed fiscal years 1986 through 1990. As cases have become more intensely and actively litigated, and the issues to be decided have grown increasingly complex, the effective management of logistics and the hearing process has become especially important. In this effort, the boards were supported by a staff which included management personnel, a legal counsel, law clerks, a librarian, legal secretaries, and docket, computer, and information specialists.

Administrative support for the boards and the panel has been automated. Systems and equipment include personal computers, displaywriters for word processing, a joint Licensing Panel and Appeal Panel library, the LEXIS automated legal research system, a docket room, and a computerized travel and timekeeping system. An internal computerized Hearing Status Report now has a virtually complete data base and is capable of generating valuable case management information.

The panel's Computer Assistance Project (CAP) to expedite large cases is well under way. Computerization of the Indian Point (N.Y.) record in 1983 proved that substantial time and labor can be saved by using a full text word search transcript. Building on that experience, the panel obtained advice, analysis, and recommendations from consultants with both legal and computer expertise. They recommended a system combining personal computers with off-the-shelf software to establish a full text word search computerized record. In place at the outset of a large case, the system would permit electronic filing, computerized transcripts, greatly expedited record searches, and faster and more complete decision-making and decision writing. By using resources for the most part already in place, the cost benefit ratio for large cases should be substantial.

The Caseload

During the fiscal year ending September 30, 1984, Licensing Boards conducted 55 proceedings involving nuclear power plants and other nuclear facilities with a construction value well in excess of $70 billion. Forty-two percent of the proceedings were completed. Some 239 days of hearings were held, comprising 174 days of trial and 65 days of prehearing conferences. Twenty-three proceedings were closed while thirteen new cases were opened. The operation of fourteen nuclear power plant units was authorized.

The mix of cases on the panel's docket has begun a shift toward smaller cases of greater diversity which is expected to continue over the next five years. In this regard the Commission ordered informal proceedings in three materials license cases to be heard by a single Administrative Judge. Two of the three judges assigned to hear the cases by the panel Chairman were technical members of the panel.

Hearing Procedure

The heavy ASLBP caseload, combined with increasing public awareness and involvement in the licensing process, has made effective hearing management essential to the timely completion of licensing decisions. Using the procedural tools available under Commission regulations, Licensing Boards have more sharply focused efforts to assure that issues for hearing are soundly based and well-defined. Prehearing conferences are utilized extensively for the purposes of reviewing and refining proposed contentions, defining the scope of relevant discovery, and developing realistic hearing schedules. The discovery process itself is closely monitored in order to eliminate unnecessary or duplicate efforts and to assure the early resolution of potentially time-consuming disputes. As a result of this active management, almost 90 per cent of the contentions filed in operating license proceedings were resolved prior to hearing. Most importantly, however, these efficiencies have been achieved through hearing management practices that insure the fundamental fairness to all parties mandated by law.

Rules of Practice

The panel's first comprehensive revision of the Commission's Rules of Practice in over a decade was submitted to the Commission in September 1984. The revision seeks to eliminate unnecessary and redundant verbiage; reorganize the regulations in a more logical order; present the rules in readable "plain English"; and incorporate procedures and standards established by NRC case law over the last 13 years which have not been codified. The Commission reviewed the panel's proposal in January 1985 and requested various refinements. Publication of the proposal in the *Federal Register* awaited Commission action at the close of the fiscal year.

Cases of Note

Fifteen operating licenses were authorized by initial decisions or orders. New units authorized included: Byron Units 1 and 2 (Ill.); Catawba Units 1 and 2 (S.C.); Clinton Units 1 and 2 (Ill.); Hope Creek (N.J.); Limerick Units 1 and 2 (Pa.); Palo Verde Units 2 and 3 (Ariz.); Perry Units 1 and 2 (Ohio); and River Bend Unit 1 (La.). The restart of Three Mile Island Unit 1 (Pa.) was also authorized. The following decisions exemplify the kinds of cases the board was involved with during the report period.

Three Mile Island Unit 1. On August 19, the Licensing Board issued its fourth and final Partial Initial Decision, closing the case. The decision addressed the "Dieckamp Mailgram" issue, i.e., whether the Licensee's Chief Executive Officer had been truthful in a mailgram of May 9, 1979, to Congressman Udall indicating that on March 28, 1979 (the first day of the Three Mile Island accident), there was no evidence that anyone interpreted the pressure spike and containment spray actuation in terms of core damage. The board found that the mailgram was accurate when sent and that there was no evidence impugning the integrity of Mr. Dieckamp.

This decision concluded a six year effort of massive proportions in which not only TMI but many aspects of the nuclear industry were examined on the record and in public. The original notice of hearing was issued in August 1979, and the staff was ready for hearing in the fall of 1980. The initial decision on management issues favorable to the utility was issued in August 1981, and the decision on all other issues (plant design, separation of Unit 1 and Unit 2, and emergency planning) was issued with conditions on December 14, 1981. That decision authorized operation at 5 percent power. However, a September 1981 board notification raised allegations of cheating on license operator examinations. That and all other remaining issues were resolved in a 1982 decision authorizing restart of the undamaged Unit 1.

Thereafter, several matters were remanded, necessitating additional hearings the last of which was the Dieckamp mailgram. During this six-year period, the presiding Licensing Boards considered virtually every aspect of public health and safety prior to authorizing restart.

Shoreham. Three major decision were issued in the Shoreham (N.Y.) proceeding. The first authorized low power testing, and the second found the Transamerica Delaval diesel generators acceptable for operation during the first fuel cycle. In the third decision, the board ruled that an operating license should not be issued for a completed plant where the utility did not have an adequate plan to respond to an emergency at the facility because State statutes prohibit the utility from activities essential to the successful implementation of the utility emergency plan. The decision was affirmed on appeal.

Palo Verde. In July 1985, the Licensing Board in Palo Verde (Ariz.) approved a Settlement Agreement reached between the joint applicants and the intervenor regarding the environmental issue raised by the latter—the asserted adverse impact that salt deposition associated with the operation of the Palo Verde facilities will have upon the productivity of nearby agricultural lands cultivated by intervenor members. The settlement agreement requires joint applicants to conduct a specific agricultural monitoring program requested by the intervenor. If harmful effects or evidence of trends toward irreversible damage are observed, a detailed analysis of the data

Atomic Safety and Licensing Boards authorized 15 operating licenses by initial decisions or orders during the report period, including one plant (Three Mile Island Unit 1) authorized to restart operation. Hearings on issues related to operation of the Perry nuclear plant in Ohio are shown in progress. Attorney Colleen Woodhead represented the NRC staff during the hearing.

will be provided together with a proposed course of action to alleviate the problem. The board's action terminated the formal adjudicatory proceeding.

The foregoing noteworthy cases dealt with operating license proceedings. Two construction permit cases were also completed during the fiscal year leaving only two cases on the docket, both in suspended status. In Fulton (Pa.) the application was dismissed without prejudice conditioned on a bar against an identical reactor being constructed at the site. The final chapter in the Clinch River (Tenn.) breeder reactor proceeding was written by a decision on site redress and dismissal of the application without prejudice.

Also of interest were two Illinois proceedings involving the Kerr-McGee Chemical Corporation, one concerning cleanup of thorium contamination and the other concerning disposal of radioactive mill tailings, the board held that the corporate applicant had not waived its right to challenge the government's disposition of its disposal application. The board found that the NRC staff's treatment of the disposal application violated the National Environmental Policy Act. The board also held that the staff may not automatically apply EPA standards to justify a clean up order. The holding required staff to show that potential health hazards justify cleanup.

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

Atomic Safety and Licensing Appeal Boards, consisting of three members each, perform review functions for the Commission in facility licensing proceedings and others the Commission may specify. Unless the Commission decides to review an Appeal Board decision, that decision becomes the final agency order and is subject only to judicial review in a Federal court of appeals. The board for each proceeding is selected from among the members of the Atomic Safety and Licensing Appeal Panel (ASLAP) by the panel chairman. (See Appendix 2 for membership of the panel.)

Under Commission rules, the Appeal Boards hear appeals from Licensing Board decisions and certain rulings on intervention petitions by members of the public seeking to become parties to the proceeding. They also review Licensing Board decisions on their own initiative in the absence of any appeal. In limited circumstances, Appeal Boards also consider questions posed by the parties or the Licensing Board, or rulings referred by a Licensing Board, while the proceeding is still in progress. Appeal Boards also occasionally conduct evidentiary hearings either as part of their appellate review function or on direction from the Commission.

During the report period, the Appeal Boards were called upon to review decisions and rulings of Licensing Boards in 15 separate licensing proceedings, resulting in more than 30 Appeal Board decisions published in the Nuclear Regulatory Commission Issuances (NRCI). (The NRCIs are a permanent collection of NRC licensing and other decisions available to the bar and the general public). These reviews also resulted in a number of other less significant unpublished memoranda and orders. Highlighted below are the more significant Appeal Board decisions.

TMI-Restart Proceeding

This proceeding to consider the restart of the Three Mile Island (Pa.) facility continued to occupy a substantial amount of Appeal Board time. Earlier, the Appeal Board had dealt with a number of matters relating to whether the plant could be operated in a safe and environmentally sound manner. This past year, the Appeal Board was presented with additional issues for resolution. A significant safety issue involved the efficacy of repairing leaking steam generator tubes by a kinetic expansion process in lieu of plugging those tubes and removing them from service. In another decision, the Appeal Board reviewed the request of intervenors to reopen the proceeding on the issue of management competency to operate the unit. The Appeal Board rejected the request finding that it had been made too late and that, in any event, the new information submitted by the intervenors in support of their request would not have produced a different result from that earlier determined. Subsequently, the Commission authorized the restart of the plant.

Diablo Canyon

In 1982, the Licensing Board had determined that the Diablo Canyon (Cal.) facility can be operated without endangering the health and safety of the public and authorized a full power license for the facility. Questions concerning the adequacy of the design of the plant subsequently arose. The Appeal Board considered those questions and ultimately endorsed the Licensing Board's safety determination with respect to Unit 1 of the plant. During the past year, the Appeal Board considered the question of the adequacy of the design of Unit 2. It found that the designs of Units 1 and 2 were virtually identical and that the results of the program undertaken to verify the adequacy of the design of Unit 1, together with other evidence, supported a similar favorable determination for Unit 2.

Shoreham, Limerick, Waterford

Three other operating license proceedings occupied a major portion of the Appeal Board's time during the course of the year. The Shoreham (N.Y.) plant was the subject of four published Appeal Board decisions, while the Limerick (Pa.) and Waterford (La.) plants were involved in six published decisions each.

One of the *Shoreham* decisions dealt with whether all of the plant's safety and safety-related structures, systems and components were constructed in accordance with the Commission's quality assurance requirements. The Licensing Board had determined that question affirmatively, and on appellate review, the Appeal Board agreed. In another *Shoreham* decision, the Appeal Board declined to stay a Licensing Board decision authorizing a low-power license for the plant. In doing so, the Appeal Board found no merit to the intervenors' claim that they would suffer irreparable injury in the absence of such a stay.

In *Limerick*, the Appeal Board was called upon to rule upon several stay requests. Two of these sought the stay of a Licensing Board decision authorizing a low-power license to the applicant after the license had already issued. In that circumstance, the Appeal Board treated them as requests for suspension of the underlying authorization for the license. Finding that the intervenors had not shown that they would prevail on the merits or would be irreparably harmed absent the stay, the board denied the request. Another request sought the stay of a later Licensing Board decision authorizing a full-power license for the plant. The Appeal Board denied the request because the intervenors had failed to carry the heavy burden of demonstrating, on the basis of the established stay criteria, that a stay was warranted. In an earlier decision involving that plant, the Appeal Board reversed a Licensing Board decision that had dismissed the inmates of the State Correctional Institution at Graterford, Pa., from the proceeding. (The Graterford institution is located eight miles from the Limerick facility and the inmates' concerns related to the emergency response plan.)

After the record had closed in *Waterford*, the intervenors made several requests to reopen the record of the proceeding. One of these sought a hearing on the proper construction of the basemat underlying the facility. After detailed consideration, the Appeal Board found the justification inadequate and denied the request. In another decision, the Appeal Board also denied another intervenor request to reopen the record—this time on the issue whether there was reasonable assurance that the Waterford facility, as built, can and will be operated without endangering the public health and safety. The Appeal Board found that the intervenors had failed to make its case for a further hearing on that subject.

Other Proceedings

Quality Assurance. In the *Byron* (Ill.) proceeding, the Licensing Board had denied the application for an operating license for the plant because of deficiencies in the applicant's quality assurance program. Of particular concern to the board was whether the inspectors were qualified to perform their functions so that significant construction defects may have gone undetected. Following an appeal by the applicant, the Appeal Board remanded the proceeding to the Licensing Board to determine whether a program calling for the reinspection of the work of these inspectors provided the requisite degree of confidence that the inspectors were competent. After taking evidence on the matter, the Licensing Board concluded that the reinspection demonstrated that the inspectors were competent and authorized the issuance of a full-power license for the plant. On appeal by the intervenors, the Licensing Board's decision was upheld by the Appeal Board.

Challenges to the applicant's quality assurance programs were also involved in the *Perry* (Ohio.) and *Catawba* (S.C.) proceedings. In each proceeding the Licensing Board had found the quality assurance program for the plant to be adequate. Subsequently, intervenors in each proceeding sought to reopen the record to receive further evidence on the issue. In each case, the Appeal Board found that the intervenors had failed to justify their request.

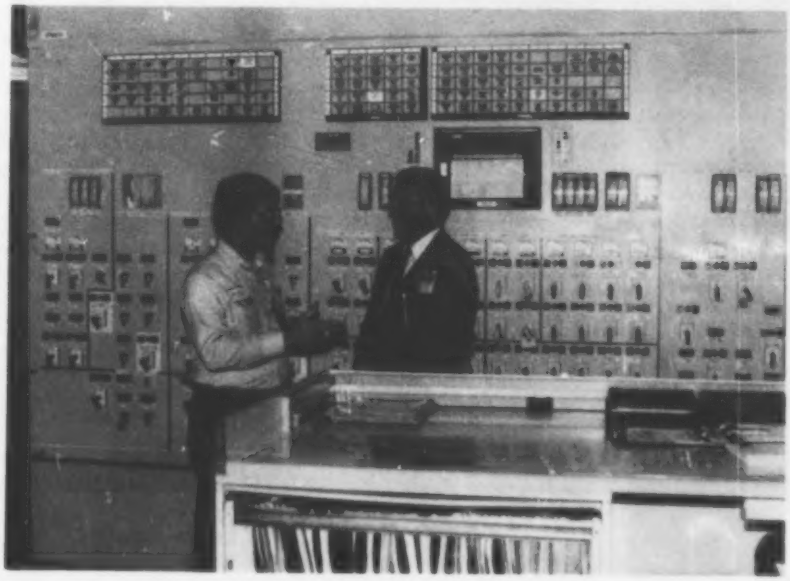
Management Competence. *South Texas* (Tex.) presented the Appeal Board with the question whether the Licensing Board had utilized the proper standard in determining that the applicant is likely to meet the character and competence requirements necessary to obtain an operating license for the plant. On this score, the Licensing Board had concluded that it was called upon to examine the applicant's record of compliance with NRC regulations, its response to ascertained deficiencies, and its candor in dealing with the Commission, the board, the staff and other parties. The intervenors disagreed with part of the standard, maintaining that remedial measures undertaken by the applicant to correct earlier deficiencies in construction or quality assurance should not be factored into whether the applicant had the necessary character and competence. Upon review of the matter, the Appeal Board agreed with the Licensing Board's approach, pointing out that the NRC's regulatory scheme recognizes that an applicant is bound to make errors necessitating correction.

Public Intervention. Challenges to Licensing Board rulings on petitions by members of the public desiring to participate in licensing proceedings were resolved by the Appeal Board in two published decisions. In *North Anna* (Va.) an intervenor appealed from a portion of a Licensing Board order in which the board denied his admittance as a party in one of two related proceedings involving proposed amendments to the North Anna operating licenses to enable that facility to receive and store spent fuel generated at the applicant's Surry (Va.) plant. The intervenor had submitted identical contentions in the second proceeding, to which he had not been admitted. In dismissing the appeal, the Appeal Board noted that, in Commission practice as in judicial proceedings, only an injured party may appeal. The intervenor had not been injured by the rejection in the second proceeding since, by reason of his admission to the first, he would be able to litigate his full range of concerns. And in *Pilgrim* (Mass.), an operating license amendment proceeding, the Appeal Board affirmed the Licensing Board's denial of an intervention petition for failure to meet the necessary admission criteria.

Other Interlocutory Appellate Review. Under Commission rules, apart from orders on intervention petitions, interlocutory rulings of Licensing Boards —i.e., rulings issued during the course of a proceeding, as contrasted with the decision at the end of the proceeding—are not immediately appealable as a matter of right. Generally speaking, the Appeal Board will review such rulings as a matter of discretion only if the Licensing Board ruling either (1) threatens the party adversely affected by it with immediate and serious irreparable impact which, as a practical matter, could not be alleviated at a later appeal, or (2) affects the basic structure of the proceeding in a pervasive or unusual manner.

In *Braidwood* (Ill.), the applicants had sought discretionary Appeal Board review of a Licensing Board decision that allowed the intervenors to amend, after discovery, a contention submitted in connection with their intervention petition, that the board had previously found to be insufficiently specific. In declining to undertake interlocutory review of the Licensing Board's ruling, the Appeal Board noted that the basic structure of an ongoing adjudication is not changed simply because the admission of a contention results from a licensing board ruling that is important or novel, or may conflict with case law, policy or Commission regulations.

Sua Sponte Review. Under Commission practice, Appeal Boards review on their own initiative (i.e., *sua sponte*) the Licensing Board decisions and the underlying record on every safety and environmental issue considered by the Licensing Board, even where no appeal has been taken on a particular issue. The Appeal Board completed its *sua sponte* review of the Licensing Board's decision in the *Wolf Creek* (Kan.) proceeding and affirmed the Licensing Board's decision authorizing an operating license for the plant.



NRC Commissioner Lando W. Zech, Jr., is shown talking with a technician during a tour of the Byron nuclear power plant near Rockford, Ill. In hearings held in 1985, the Licensing Board reversed an earlier decision denying an operating license and authorized issuance of a full-power license for Byron Units 1 and 2.

COMMISSION DECISIONS

Some of the Commission's more significant decisions during fiscal year 1985 are discussed below. The Commission's actions on export licensing cases are discussed in Chapter 10.

Reconsideration Procedures Unavailable to "Non-Parties"

In *General Public Utilities Nuclear Corporation* (Three Mile Island Nuclear Station, Units 1 and 2) (Oyster Creek Nuclear Generating Station), CLI-85-4, 21 NRC 561 (1985), the Commission made clear that the rule prohibiting individuals from using petitions under 10 CFR 2.206 as a device to obtain agency reconsideration of a matter previously decided by a Licensing Board, or a different administrative forum than a Licensing Board, applied to both parties and non-parties to the Commission's licensing proceedings.

This case arose when a group of petitioners who were not parties to the TMI-1 restart proceeding joined with one party to that proceeding in filing a petition under 10 CFR 2.206 requesting that the licenses to operate the TMI and Oyster Creek facilities be revoked because the licensee lacked the necessary character to safely operate the facilities, and arguing that the overall character of the licensee was not an issue in the TMI-1 restart proceeding. The Director, Office of Nuclear Reactor Regulation, denied the petition as insufficient to support an enforcement proceeding. The Commission took review of the Director's Decision to affirm but clarify the basis for the denial.

As explained by the Commission, the firmly established "reconsideration/avoidance of proper forum" rule applicable to parties in a licensing proceeding who file concurrent petitions for enforcement under 10 CFR 2.206 was grounded on concerns for administrative resources and economy, and the need for finality to administrative decisionmaking. The Commission then concluded that these reasons applied with equal force to petitioners who, failing to become parties to a proceeding wherein their concerns could more properly be considered, seek to use the 2.206 procedures as a means to reopen issues previously adjudicated in that proceeding. Moreover, while the petitioners were correct that not every past issue bearing on the licensee's character would be separately adjudicated, the licensee's overall character was being considered in the TMI-1 restart proceeding to the extent those character issues were deemed significant enough to reopen the record in the proceeding. In the Commission's view, the petition failed to raise any new character issues that warranted the initiation of an enforcement proceeding.

Parties' Right to an Impartial Adjudication Does Not Include a Right to Choose the Judge

In *Metropolitan Edison Company, et al.* (Three Mile Island Nuclear Station, Unit 1), CLI-85-5, 21 NRC 566 (1985), intervenors appealed a Licensing Board member's decision declining to disqualify himself from the TMI-1 proceeding. The intervenors' disqualification motion was based on the board member's (1) letter as an individual citizen to a Federal District Court judge urging leniency in sentencing a TMI-2 employee in a related criminal matter, (2) comments during the proceeding on the treatment of three individuals, and (3) treatment of one intervenor's counsel and witnesses. The Commission affirmed the Licensing Board member's decision. In doing so, the Commission clarified the scope of a party's right to an impartial judge.

In the Commission's view, a party has no right to expect that favorable rulings will be divided equally between the parties, or that a judge may not occasionally use strong language toward a party or in expressing his views on matters before him. More-

The Wolf Creek nuclear power plant was licensed to operate in 1985. It is owned by the Kansas Gas and Electric Company, Kansas City Power and Light Company and the Kansas Electric Power Cooperative, Inc. The facility is located in Coffee County, Kans., 3.5 miles northeast of Burlington, Kans.

over, the Commission noted that the fact that a judge's actions might be controversial or may provoke strong reactions by the parties are not, standing alone, grounds for disqualification. A party does not have a right to the judge of his or her choice. Rather, a party to an adjudicatory proceeding has a right to an impartial adjudicator, both in reality and in appearance to a reasonable observer.

As to the Licensing Board member's comments and treatment of the intervenors, the Commission summarily sustained the member's decision. As to the letter to the Federal District Court judge, the Commission concluded that the member's action did not violate any ethical or professional responsibility standard, and were not otherwise grounds for disqualification since the member had made clear in the letter and his decision that the statements made in the letter reflected the member's personal opinion based solely on the administrative record in and conduct of the TMI-1 proceeding.

Shutdown or Additional Remedial Safety Measures Unnecessary at Indian Point Facility

In May 1980, the Commission issued an order (unpublished) announcing an intent to initiate a discretionary adjudication for the purpose of resolving safety issues concerning Indian Point raised in a petition filed by the Union of Concerned Scientists. In *Consolidated Edison Company of New York* (Indian Point, Unit No. 2) and *Power Authority of the State of New York* (Indian Point, Unit No. 3), CLI-85-6, 21 NRC 1043 (1985), the Commission concluded that neither a shutdown of the facility nor the imposition of additional remedial actions beyond those already implemented by the licensees was warranted in light of information developed during and after the special proceeding.

While recognizing that there can be no truly reliable quantitative comparison of risks among nuclear power plants, the Commission concluded that Indian Point was not a risk "outlier" (i.e., in a higher risk class all its own) and thus, did not present a risk to the public significantly greater than that imposed by other NRC-licensed plants. The Commission's conclusions were based primarily on engineering judgment of plant safety as demonstrated by a thorough probing of the Indian Point units and by evaluation of the risk reduction effectiveness of plant safety systems. A secondary factor was the fact that the quantitative risk assessments adopted by the board indicated that the level of risk was acceptably low. Finally, the Commission directed the staff to investigate wind vulnerability at the facility, to keep abreast of filtered vented containment research and experience, and to report within 60 days to the Commission on the status of emergency planning at Indian Point, which had been inadequate at the end of the special proceeding hearings but had improved prior to the Commission's decision.

TMI-1 Granted Conditional Authority To Resume Operations

Following the March 28, 1979 accident at TMI-2, the Commission issued an immediately effective enforcement order (unpublished) directing that TMI-1, which had been routinely shut down on February 15, 1979 for refueling, remain shut down until further order. In a subsequent published decision, the Commission established the restart proceedings and noted that, since the law required the immediate effectiveness aspect of its enforcement order to be lifted once the basis for that action was adequately resolved, any subsequent "effectiveness review" by the Commission would be considered apart from the normal administrative appellate review applicable to the merits of the enforcement order. *Metropolitan Edison Company* (Three Mile Island Nuclear Station, Unit No. 1), CLI-79-8, 10 NRC 141, 149 (1979).

The formal adjudicatory record was closed in this proceeding in 1981.

In CLI-85-2, 21 NRC 282 (1985), the Commission decided to permit, as a matter of discretion rather than law, the Licensing Board to render decisions on two issues (adequacy of training at TMI-1 and the Dieckamp mailgram) where the Licensing Board had already completed hearings pursuant to the Appeal Board decision then under Commission review. In addition, the Commission concluded that since the individuals involved in pre-accident leak rate falsifications (Hartman Allegations) were no longer involved in the day-to-day operation of TMI-1, the allegations presented no significant safety issue concerning the present management of TMI-1. Nonetheless, the Commission gave notice that it intended to initiate a separate hearing to determine the ultimate status of those involved. As to any other issue, the Commission found that no further hearings were warranted within the restart proceeding since no issue met the standards for reopening, i.e., raised a significant safety concern which might have affected the Licensing Board's decision.

In CLI-85-9, 21 NRC 1118 (1985), the Commission concluded, based on the formal record created by the restart proceeding, that (1) the concerns which led the Commission to make the 1979 order immediately effective had been adequately resolved, and (2) the two remaining issues (adequacy training and Dieckamp mailgram) had no bearing on the Commission's ability to make its "effectiveness" determination. Thus it felt legally obligated to lift the immediate effectiveness of its 1979 order notwithstanding the pendency of further appellate review. The Commission noted that the concerns about the licensee training program that had led to the 1979 order were based on the licensee's pre-accident program. Since the accident, the training program was the subject of favorable Licensing Board findings and bore little resemblance to the pre-accident program, and there were no present concerns relative to training at TMI-1 that warranted a continuation of the immediate effectiveness portion of its 1979 order. As to the Dieckamp mailgram matter, the Commission found that, since

The five members of the Nuclear Regulatory Commission are shown in an open meeting during 1985. On the far side of the conference table, left to right, are: Commissioner Lando W. Zech, Jr., Commissioner Thomas M. Roberts, Chairman Nunzio J. Palladino, Commissioner Frederick M. Bernthal, and Commissioner James K. Asselstine.

Mr. Dieckamp was no longer involved in the day-to-day operations of TMI-1, any inquiry into the statements made in the mailgram would not raise health and safety concerns justifying an immediately effective shutdown order. The Commission further rejected intervenor attempts to characterize the restart proceeding as a license amendment proceeding rather than an enforcement proceeding simply because the Licensing Board had imposed conditions on the license. The Commission went on to note that no findings adverse to the licensee had been made with respect to the Dieckamp mailgram. Finally, the Commission found that special restart conditions were not justified due to assorted management integrity concerns. Rather, such conditions were warranted because the facility and its various safety systems had been shut down for six years. The Commission required conditions included the submission and approval of a proposed power ascension schedule, increased NRC monitoring during the initial restart process, and in-depth evaluations of the facility, its operation and its management at the end of six months and twelve months. The Commission's action was judicially in *Three Mile Island Alert, Inc. v. NRC* (3d Cir. No. 85-3001) and related cases discussed below.

San Onofre Restart Order Not a License Amendment

In *Southern California Edison Company, et al.* (San Onofre Nuclear Generating Station, Unit 1), CLI-85-10, 21 NRC 1569 (1985), the Commission held that an order which rescinded, in whole or in part, a prior order which was not itself a formal license amendment need not be treated procedurally as a license amendment where the effect of the rescission letter was to restore to the licensee the authority to proceed under the original license. Thus, interested persons are not, as a matter of law, entitled to a hearing on the merits of the later order.

The case had its origins in 1982 when the licensee, faced with NRC concerns that the facility might not meet its original 0.5g design basis seismic criterion due to high stresses reported in certain piping systems and mechanical equipment, committed to resolve the concerns by upgrading Unit 1 to meet the 0.67g seismic criterion applied to Units 2 and 3 and to maintaining the facility in a shutdown condition until completion of the upgrade program.

The NRC responded to the licensee's proposal by issuing a confirming order requiring the licensee to maintain Unit 1 in a shutdown condition until upgrade modifications were completed and approved. Based on a subsequent licensee request that operation of Unit 1 be authorized prior to final completion of the upgrade program, the NRC issued an order in November 1984 rescinding the 1982 order but conditioning any restart on the expeditious completion of the upgrade program. Based on their position that the rescission order was in essence a license amendment, petitioners requested a hearing on the merits of the November 1984 order and a stay of that order pending completion of the hearing.

The Commission denied the request. In the Commission's view, the 1982 order neither expanded the licensee's authority under its 1981 operating license nor did it authorize or direct the licensee to take actions inconsistent with its existing license. Instead, the 1982 order cut back on the licensee's authority and was, in effect, a license suspension, an action which is legally distinct from a license amendment. This being true, the Commission had the discretionary enforcement authority to relax or modify the prior suspension order, or further condition the licensee's ability to exercise its right under the existing license short of formally amending the license. The Commission also denied the petitioners' stay request, characterizing the request as a petition for enforcement action under 10 CFR 2.206. The Commission pointed out that the effect of a stay would be the shutdown of the plant, and since such action was warranted under 2.206 only in the case of a willful violation or an immediate threat to the public health and safety, the absence of these factors in the instant case was fatal to the request. In subsequent judicial actions, the Ninth Circuit Court of Appeals similarly denied petitioners' request for relief.

Shoreham Low Power License Unaffected by Uncertainties Over Full Power Operation

In a series of orders, the Commission permitted fuel loading, precritical and cold critical testing, and low power testing at the Shoreham Nuclear Power Station, Unit 1, and then upheld the effectiveness of the low power license in the face of uncertainties as to the ultimate grant of a full power operation license for the facility.

In *Long Island Lighting Company* (Shoreham Nuclear Power Station, Unit 1), CLI-84-21, 20 NRC 1437 (1984), the Commission made effective a Licensing Board order authorizing fuel loading and precritical and cold critical testing despite the absence of qualified emergency AC power system but made its ruling subject to future favorable resolutions of deficiencies in the existing record regarding studies on safety equipment control system failures, the applicant's response to identified problems in construction cleanup, and environmental qualification of electrical equipment. In taking this action, the Commission concluded that, since not all health and safety requirements logically apply to some or all phases of low power operation, the need to satisfy a regulation prior to the issuance of a low power license must be based on a determination whether the purpose of the particular regulation has any application to the activities authorized. In the case then before the Commission, it agreed with the Licensing Board that the failure to comply with 10 CFR Part 50, Appendix A, General Design Criterion 17 (AC power) did not prevent the authorization of fuel loading and precritical and cold critical testing since the purpose of that

The Shoreham nuclear power plant in Suffolk County, N.Y., remained the subject of repeated hearings because of challenges involving emergency planning, design criteria and cost-benefit factors, and other issues. At year's end, the NRC had permitted fuel-loading, pre-critical, cold critical and low-power testing at the plant.

requirement—ensure that there is sufficient AC power to provide core cooling in the event of a postulated accident—was inapplicable to any accident that could reasonably result from fuel loading and precritical and cold critical testing.

In CLI-85-1, 21 NRC 275 (1985), the Commission made effective a Licensing Board decision granting the applicant an exemption from the requirements of General Design Criterion 17 with respect to Phase III (1% rated power) and IV (1-5% rated power) of the Shoreham low power testing program. In doing so, the Commission relied upon the board finding that the applicant's alternate AC power system provided sufficient redundancy, capacity, testability, and reliability to provide reasonable assurance of safety for low power operation of the Shoreham facility. Moreover, the Commission declined to infect its low power decisions with speculations over whether a full power license will ever be granted to the facility in light of emergency planning uncertainties, noting that the Commission's authority to issue a low power license is not dependent on a predictive finding of reasonable assurance that a full power license will eventually be issued.

Finally, in CLI-85-12, 21 NRC 1587 (1985), the Commission denied a motion by the State of New York and Suffolk County requesting a supplemental environmental impact statement (EIS) separately assessing the costs and benefits of low power testing at Shoreham in light of the assumption that no full power license will be granted for the Shoreham facility. The Commission held that uncertainties over pending full power issues (e.g., off-site emergency planning) did not mandate a supplemental EIS or some renewed cost/benefit analysis. The Commission went on to note, however, that given the valuable benefits derived from low power testing (e.g., the early identification and correction of plant problems) and the small, well-known environmental effects of low power testing, it would find that the safety benefits outweighed the environmental costs even at the low power stage.

JUDICIAL REVIEW

The more significant litigation involving the Commission either resolved during fiscal year 1985 or pending at the close of the fiscal year is summarized below.

Pending Cases

Coalition for the Environment, St. Louis Region, et al. v. NRC, et al. (D.C. Cir. No. 84-1313).

New England Coalition on Nuclear Pollution, et al. v. NRC (D.C. Cir. No. 84-1514).

In New England Coalition, the petitioner is challenging the Commission's latest amendment of its financial qualifications rule (10 CFR 50.33(f)) which, as amended, exempts from consideration in the Commission's licensing proceedings any consideration of the financial qualifications of rate-regulated utilities applying for power reactor operating licenses. The Commission did retain financial qualification for such applicants seeking construction permits. In Coalition for the Environment, petitioners challenged the Commission's issuance of a low-power license in the Callaway proceeding without considering and adjudicating the utility-applicant's financial qualifications. In January 1985, the D.C. Circuit consolidated these cases. All briefs filed by the beginning of April 1985 and oral arguments were heard on October 11, 1985.

Cuomo, et al. v. NRC (D.C. Cir. No. 85-1042).

In this action, the State of New York and Suffolk County, N.Y., are challenging the issuance of a low-power license for the Shoreham facility located in Suffolk County. In support of their challenge, the petitioners argue: (1) that their refusal to cooperate with the utility-applicant's full-power emergency plan makes the grant of a full-power license so uncertain that a recalculation of the cost/benefit equation under the National Environmental Policy Act is required, and (2) the Chairman improperly refused to recuse himself from the Shoreham proceeding in light of alleged ex parte contacts with some of the parties to the Shoreham proceeding. The Chairman had previously denied petitioner's administrative motion for recusal as factually unsupported and inexcusably late. In July 1985, the D.C. Circuit declined to stay the Commission's order authorizing low-power operation of the Shoreham facility. Oral argument is scheduled for January 1986 although the parties have requested that oral arguments be postponed to permit a full briefing of all the disputed issues related to the low-power license.

General Electric v. NRC (D.C. Cir. No. 80-2496)
Prairie Alliance v. NRC (C.D. Ill. No. 80-2095)
General Electric v. NRC (C.D. Ill. No. 80-2244)
General Electric v. NRC (7th Cir. No. 84-2066)

On May 7, 1980 the Prairie Alliance sued the NRC under the Freedom of Information Act (FOIA) to compel disclosure of the General Electric Nuclear Reactor Study known as the Reed Report. While that lawsuit was pending, the Commission, on a 2-2 vote, was unable to muster a majority vote necessary to assert any FOIA exemption protecting the report and hence ordered its release. The General Electric Company sued to enjoin release of the report and to require its return to General Electric. In 1983, the District Court granted the government's motion for summary judgment holding that the Reed Report was an "agency record" subject to the FOIA and the NRC did not abuse its discretion in releasing the Reed Report. GE appealed to the Seventh Circuit.

In a decision issued on December 21, 1984, the Seventh Circuit generally sustained the NRC's action but remanded the case to the agency for a more expansive statement of reasons for its decision to release the Reed Report. Pursuant to a May 13, 1985 letter from the Commission, on July 1, 1985 GE submitted further information regarding whether the report should be released. After those requesting the Reed Report have had an opportunity to respond to GE's submission, the Commission will then have to determine the appropriate disposition of the Reed Report.

Florida Power and Light Company v. Lorion, U.S., 84 L. Ed.2d 643 (1985).

In February 1982, Joette Lorion sought review by the D.C. Circuit Court of Appeals of the NRC's decision denying her request that Turkey Point, Unit 4, be shut down for a steam generator inspection, alleging that the Commission acted unlawfully (1) in treating her letter requesting such action as a petition under 10 CFR 2.206 and (2) in denying her request. In July 1983, the D.C. Circuit Court upheld the NRC's action in treating Lorion's letter under 10 CFR 2.206 but held sua sponte that the courts of appeals lack subject matter jurisdiction to review denials by the Nuclear Regulatory Commission of requests under 10 CFR 2.206 for enforcement action against NRC licensees (712 F.2d 1472). The court stated that jurisdiction to review such denials lies initially in the district court. On appeal, the Supreme Court reversed the D.C. Circuit and held that the courts of appeal do have such jurisdiction. However, the Court expressed no views on the merits of Lorion's petition and remanded the case back to the D.C. Circuit for further proceedings. Oral argument on the merits of Lorion's petition were held in December 1985.

Oystershell Alliance, et al. v.NRC, et al. (D.C. Cir. No. 85-1182.)

In March 1985, the Oystershell Alliance and others filed a petition for review in the U.S. Court of Appeals for the D.C. Circuit. The petition challenged the Commission's authorization of full-power operation of Waterford 3 while two motions to reopen the record were still pending before the Appeal Board. In April, the D.C. Circuit denied petitioners' request for emergency stay. The Court through Judges Wald and McGowan, found that "[t]he balance of equities does not favor the grant of a stay." The Court also stated that it "assumes that the agency's Appeal Board will act expeditiously in resolving petitioners' two motions to reopen " Subsequent to the denial of the stay, both motions to reopen were denied. Although the cases has been briefed, no oral argument has been scheduled by the D.C. Circuit.

San Luis Obispo Mothers for Peace, et al. v. NRC (D.C. Cir. Nos. 81-2035, 83-1037, 84-1042, 84-1410).

In these consolidated cases, petitioners are challenging the Commission grant of both the low-power and full-power licenses authorizing the operation of the Diablo Canyon Unit 1 facility. Following the issuance of the full-power license in August 1984, the petitioners obtained a stay of the order from the D.C. Circuit. In December 1984, the D.C. Circuit affirmed the Commission's issuance of a full-power license but held that the Commission had erred in extending the term of the low-power license without first holding an adjudicatory hearing (751 F.2d 1287). However, because the full-power license had already been issued, the court found that the error did not warrant redress. On consideration en banc, the D.C. Circuit vacated that portion of the decision dealing with earthquakes and emergency planning and ordered those issues to be re-argued. Oral argument on the en banc reconsideration of the earthquake and emergency planning issues was held October 3, 1985.

Union of Concerned Scientists v. NRC (D.C. Circuit No. 84-1549).

In this action, the Union of Concerned Scientists (UCS) sought review of the Commission's issuance of a final rule which deleted from nuclear reactor operating licenses the June 30, 1982 deadline for documentation and completion of environmental qualification of safety-related equipment. UCS argued that the NRC unlawfully deleted the deadline in violation of the Atomic Energy Act and the Administrative Procedure Act and requested the court to declare the Commission's Final Rule to be null and void and to reinstate the June 30, 1982 compliance deadline. All briefs were filed in this action by May 1985 with oral argument anticipated during the beginning of fiscal year 1986.

Resolved Cases

Cranston, et al. v. Reagan, et al. (D.D.C. Civil Action No. 84-1545).

In this action against the President, the Secretaries of State and Energy, the Director of the Arms Control and Disarmament Agency and the five NRC Commissioners, three members of Congress and six environmental groups challenged defendants' approval and implementation of certain "Agreed Minutes" to the Agreements for Cooperation with Sweden and Norway. Plaintiffs claimed that the provisions of the minutes, which provide for the advance, long-term consent of the United States to the transfer of spent reactor fuel subject to the Agreements to France and the United Kingdom for purposes of reprocessing, violate the Nuclear Nonproliferation Act. Plaintiffs argue that approval of reprocessing can only be done on a case-by-case basis. In a decision filed June 20, 1985, the District Court dismissed the action as raising a non-justicable political question, concluding that the possible consequences of judicial action would inappropriately interject the court into the President's constitutional authority over foreign affairs.

Duke Power Company v. NRC (4th Cir. No. 84-1866).

In this case, the Duke Power Company challenged as arbitrary and capricious the Commission's approval of a staff decision declining to grant the company a regulatory exemption which would have permitted the Emergency Operations Facility for the Oconee Nuclear Station to be located 125 miles from the plant. In support of its challenge, Duke Power argued that the public's health and safety would be enhanced by allowing the utility to use, during an accident, its corporate headquarters as its Emergency Operations Facility. In June 1985, the Fourth Circuit affirmed the Commission's decision. In its per curiam decision, the Fourth Circuit noted its unwillingness to substitute its judgment for that of the Commission, deferring to the Commission's responsibility and expertise in the field of nuclear safety.

Joseph W. Johnston v. NRC, et al. (7th Cir. No. 84-1583) (On appeal from N.D. Ill. No. 83-C-3615) Rockford Newspapers, Inc. v. NRC, et al. (N.D. Ill. No. 83-C-20074).

In August 1983, the American Civil Liberties Union sought a declaratory judgment that the Government in the Sunshine Act, 5 U.S.C. 552b, applies to proceedings before NRC Licensing Boards. In November 1983, the Government requested the District Court to dismiss the case on the ground that the particular acts complained of (i.e., in camera, ex parte hearings of the Licensing Board with confidential informants) did not take place, or, in the alternative, grant summary judgment to the Government. In February 1984, the District Court granted the summary judgment in the Government's favor concluding that the Sunshine Act did not apply to the Commission's Licensing Board. In July 1985, the Seventh Circuit dismissed the appeal for want of jurisdiction. Viewing the plaintiff's action as an attempt to challenge a nonexistent meeting, the Seventh Circuit concluded that the District Court lacked the subject matter jurisdiction to originally hear the case.

Three Mile Island Alert, Inc., et al v. NRC, et al. (3d Cir. No. 85-3001)

Commonwealth of Pennsylvania v. NRC, et al. (3d Cir. No. 85-3302)

Union of Concerned Scientists v. NRC, et al. (3d Cir. No. 85-3310)

Aamodt, Norman, et al v. NRC (3d Cir. No. 85-3315)

Three Mile Island Alert v. NRC, No. A-235 (Supreme Court).

In these four consolidated cases, petitioners challenged the Commission's May 1985 decision authorizing the restart of Three Mile Island Unit 1. On June 7, 1985 the Court granted petitioners' motion to stay the Commission's decision and ordered an expedited briefing schedule. Oral argument were held in June 1985. In August 1985, the court issued a decision affirming the Commission's restart Order. However, after the Commission informed the court that it intended to authorize restart, the court issued an order staying restart until the court had an opportunity to act upon any petitions for rehearing en banc.

In September, the full Third Circuit voted 10-2 to deny the four petitions to review the panel decision en banc. However, the Third Circuit extended the stay so that petitioners could to attempt to seek a stay pending Supreme Court review.

TMIA, supported by the Commonwealth of Pennsylvania, UCS, and the Aamodts, sought a stay from Justice Brennan. Justice Brennan issued a housekeeping stay to consider the request. The United States opposed the grant of a stay. On October 2, 1985, the full Court denied the stay request by a 8-1 vote. No party filed a petition for writ of certiorari, thereby ending this litigation.

Union of Concerned Scientists v. NRC (D.C. Cir. No. 82-2053).

In September 1982, the Union of Concerned Scientists (UCS) challenged NRC's July 1982 amendments to the emergency planning rules permitting (1) issuance of initial licensing decisions without the results of preparedness exercises and (2) staff authorization of low-power operating licenses without any review of off-site emergency preparedness (47 Fed. Reg. 30232 (July 13, 1982)). In May 1984, the D.C. Circuit Court of Appeals vacated NRC's July 1982 amendments to the emergency planning rules. The court ruled that the Atomic Energy Act does not permit the Commission to exclude the results of emergency preparedness exercises from operating license hearings. The D.C. Circuit subsequently denied the NRC petition for rehearing and suggestion for rehearing en banc. In October 1984, the intervenor utilities sought appeal to the Supreme Court. In January 1985, the Court declined to take review. In a related matter, the D.C. Circuit denied petitioner's request for attorney's fees, finding that the Commission was substantially justified in defending the case.

General Public Utilities Corp., et al. v. United States (E.D. Pa. No. 81-4950); 3d Cir. No. 83-1017; S. Ct. No. 84-790.

In December 1981, the owners and operators of the Three Mile Island Unit 2 nuclear facility sued the United States, alleging damages in excess of $4 billion resulting from the accident at the facility. Plaintiff's theories of liability were that the United States, in its role as a regulator, violated statutory, regulatory or other self-imposed requirements, and failed to warn the licensee of defects in its equipment, analyses, procedures and training. Alternatively, the plaintiffs argued that the United States failed to direct the licensee to correct deficiencies contributing to the accident. In November 1982, the District Court denied the Government's motion to dismiss this case on both the discretionary function and the misrepresentation exemptions to the Tort Claims Act. However, recognizing that these issues were close and important, the District Court certified an immediate appeal to the Third Circuit. In September 1984, the Third Circuit reversed the District Court and ordered the case dismissed on discretionary function grounds. In February 1985, the Supreme Court declined to take review.

Guard v. NRC (D.C. Cir. No. 84-1091).

In this case, originally filed with the Ninth Circuit Court of Appeals but later transferred to the D.C. Circuit, Guard challenged the issuance of a full-power license for operation of the San Onofre Unit 3 facility which also deleted a condition to the operating license regarding off-site medical service arrangements. That condition on operation was attached to the low-power license issued on November 12, 1982, and was subsequently deleted in the full-power license in accordance with a Licensing board decision of August 12, 1983, which found that the arrangements for off-site medical services were consistent with 10 CFR 50.47(b)(12), as interpreted by the Commission in CLI-83-10, 17 NRC 528 (1983). The petition requested that the court review and set aside that part of the order authorizing the deletion of the condition. In February 1985, a unanimous panel of the D.C. Circuit vacated the Commission's generic interpretation of the emergency planning standard in 10 CFR 50.47(b)(12) and held that the requirement for "arrangements" for "contaminated injured individuals" demanded something more than a mere listing of area hospitals which could treat such persons.

Management and Communication

CHAPTER 13

Progress on Consolidation

Since its inception, the NRC has sought a remedy to the broad dispersion of its Headquarters staff in various venues in and around the Washington, D.C. area. The Congress, the Government Accounting Office, the Office of Management and Budget (OMB), and a number of study commissions have stressed that its multiple office locations have impeded NRC's ability to accomplish its mission. Past efforts to bring about a consolidation of staff offices have fallen short of success.

Currently, the Administrator of the Government Services Administration (GSA) has indicated a preference for pursuing NRC consolidation through the purchase of one or more buildings. The interest of the real estate developers and building owners in selling properties to the Government has been solicited. The GSA, NRC, and the Subcommittee on Public Buildings and Grounds have also explored other options. The Commission had determined, at the close of the report period, that the purchase of an acceptable building(s) in suburban Montgomery County (Md.) would be acceptable. The GSA was pursuing a purchase option on a specific building at that time. The preferred site includes a single building, which could house approximately one-half of the Headquarters' operation. That would mean that six buildings currently assigned to NRC in the District of Columbia, and in Silver Spring and Bethesda in Maryland, could be relinquished. With the purchase of the building and site, sufficient land would acquired to construct a second building adequate to house the remainder of the agency.

The GSA has presented the proposed purchase plan to OMB for their review as to funding availability. Contingent on OMB approval, GSA will commence final negotiations and proceed to purchase the building. Initial occupancy could begin as early as July 1986. Actual timing of the moves will be based on final approval and purchase of the building.

STRENGTH AND STRUCTURE

Changes Within the Commission and Senior Staff

The following changes occurred on the Commission and at senior staff level during the report period:

In July 1985 Commissioner Thomas M. Roberts was reappointed to the Commission for a second term.

In January 1985, James M. Taylor was appointed Director, Office Inspection and Enforcement, succeeding Richard C. DeYoung.

In February 1985, J. Nelson Grace was appointed Regional Administrator of the Region II Office, Atlanta, Georgia, succeeding James P. O'Reilly.

In July 1985, Ronald M. Scroggins was appointed Controller/Director, Office of Resource Management, succeeding Learned W. Barry.

Personnel Management

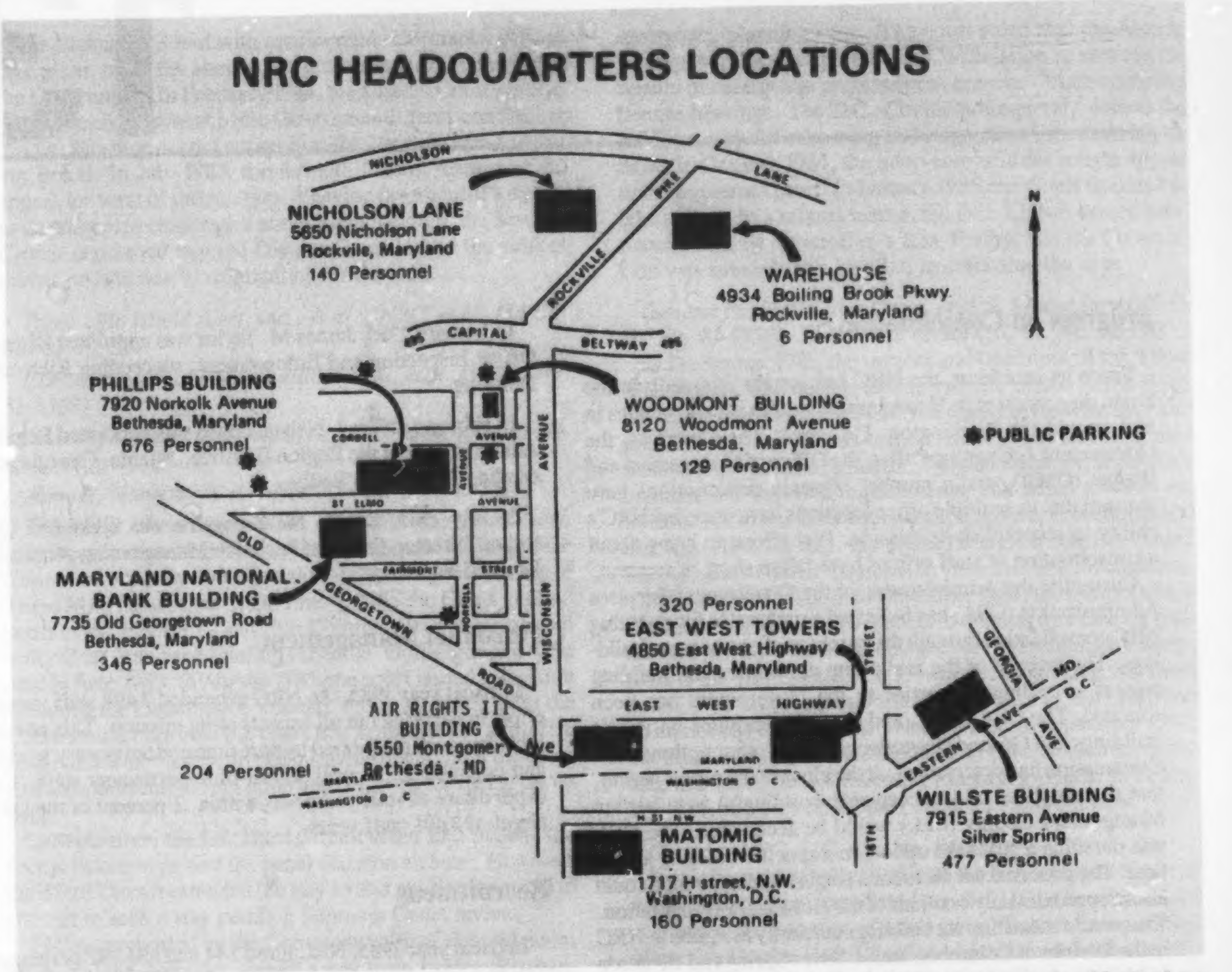
In fiscal year 1985, the NRC expended 3,498 staff years of effort in carrying out all aspects of its mission. This number includes work performed by part-time and temporary workers and consultants, as well as full-time permanent staff. Total expenditure of staff years was within .2 percent of the OMB target of 3,491 staff years.

Recruitment

In fiscal year 1985, NRC hired 244 and lost 292 permanent full-time employees, for an attrition rate of 8.7 percent per year. The Agency's recruitment program included visits to 27 college campuses (including campus "job fairs") and participation in approximately 26 other job fairs during the year. Twenty-nine of the new employees were entry-level scientists and engineers.

ORGANIZATIONAL CHANGES

The Office of Nuclear Reactor Regulation was reorganized into five new divisions to respond to the continuing shift in the Agency's workload from licensing reviews to operational safety of operating reactors. Three of the new divisions are dedicated to work related to reactors from specific vendors; one for pressurized water reactors designed by Westinghouse, one for boiling water reactors designed by General Electric, and one for pressurized reactors designed by Combustion Engineering or Babcock and Wilson and for special projects such as the Integrated Assessment Program, TMI-2 Cleanup and non-power reactor licensing. The remaining two divisions deal with safety, operations and human factors technology. (See Chapter 2.)


NRC HEADQUARTERS LOCATIONS
NICHOLSON
NICHOLSON LANE
5650 Nicholson Lane
Rockvile, Maryland
140 Personnel
PIKE
LANE
ROCKVILLE
N
WAREHOUSE
4934 Boiling Brook Pkwy
Rockville, Maryland
6 Personnel
CAPITAL
BELTWAY 495
PHILLIPS BUILDING
7920 Norkolk Avenue
Bethesda, Maryland
676 Personnel
WOODMONT BUILDING
8120 Woodmont Avenue
Bethesda, Maryland
129 Personnel
PUBLIC PARKING
CORDELL
AVENUE
ST ELMO
AVENUE
WISCONSIN AVENUE
OLD GEORGETOWN ROAD
FAIRMONT
STREET
NORFOLK
MARYLAND NATIONAL BANK BUILDING
7735 Old Georgetown Road
Bethesda, Maryland
346 Personnel
320 Personnel
EAST WEST TOWERS
4850 East West Highway
Bethesda, Maryland
EAST WEST HIGHWAY
STREET
GEORGIA AVE
MD.
D.C.
EASTERN AVE
AIR RIGHTS III BUILDING
4550 Montgomery Ave
Bethesda, MD
204 Personnel
MARYLAND
WASHINGTON D C
H ST NW
MATOMIC BUILDING
1717 H street, N.W.
Washington, D.C.
160 Personnel
16TH
WILLSTE BUILDING
7915 Eastern Avenue
Silver Spring
477 Personnel

This diagram indicates the widely dispersed venues of the various NRC offices as of 1985. After many years of effort by the Commission, the Congress and various other elements of the Federal Government, it appeared at year's end that consolidation of the agency's offices could be realized. A building complex in the Rockville area of suburban Maryland was selected as the new site for the NRC, and negotiation between the building developers and the General Services Administration was under way, subject to approval by the Office of Management and Budget.

In the Office of Nuclear Materials Safety and Safeguards, the Division of Safeguards was reorganized into three branches rather than five to carry out its program function more efficiently. The three new branches are titled Facility Assessment and Standardization, Safeguards Reactor and Transportation Licensing, and Safeguards Material Licensing and International Activities.

The Office of Inspection and Enforcement was reorganized to provide increased management attention to quality assurance, vendor, and inspection programs. The former Division of Quality Assurance, Safeguards and Inspection Programs was split into the Division of Inspection Programs and the Division of Quality Assurance, Vendor, and Technical Training Programs. The Division of Emergency Preparedness and Engineering Response remained unchanged. (See Chapter 8.)

EMPLOYEE-MANAGEMENT RELATIONS

Incentive Awards

NRC managers recognized high quality work performed by staff members during 1985 with 241 special achievement awards, 355 high quality performance increases, 117 certificates of appreciation, 70 SES bonuses, 7 distinguished service awards, and 2 equal employment opportunity awards.

Three NRC executives received Presidential Rank Awards. John G. Davis, Director of the Office of Nuclear Material Safety and Safeguards, received the Distinguished Executive Rank Award. Martin G. Malsch, Deputy General Counsel of the Office of the General Counsel and Patricia G. Norry, Director of the Office of Administration, received the rank of Meritorious Executive.

Walter S. Schwink, Senior Program Manager, Office of the Executive Director for Operations, received a Presidential Letter of Commendation.

The Commission established two new awards to recognize resident inspectors for meritorious and outstanding achievements: NRC Resident Inspector of the Year, and Regional Resident Inspector of the Year. This year's NRC Resident Inspector of the Year was Antone C. Cerne, the senior resident inspector at the Seabrook facility (N.H.) Regional Resident Inspector of the Year awards went to Thomas P. Gwynn at the Clinton plant (Ill.), Marvin M. Mendonca at Diablo Canyon (Cal.), Milton B. Shymlock at Watts Bar (Tenn.), and Lawrence A. Yandell at Fort Calhoun (Neb.).

Labor Relations

The collective bargaining agreement between the NRC and the National Treasury Employees Union was being renegotiated at the close of the report period. The ground rules for the negotiations were agreed upon in May 1985, and the comprehensive negotiations began in June 1985.

The parties agreed to allow 40 days for negotiations before seeking the assistance of the Federal Mediation and Conciliation Service. While the parties have reached agreement on numerous subordinate issues, the major issues remain unresolved.

Approximately 50 grievances, 35 mid-contract negotiations, 70 performance/ conduct actions and 13 unfair labor practice charges were handled during fiscal year 1985.

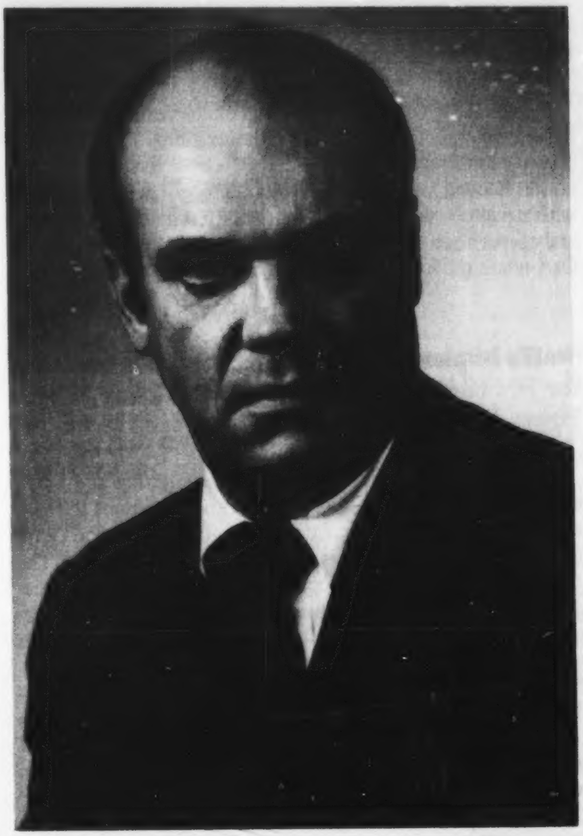
INSPECTION AND AUDIT

The activities of the NRC's Office of Inspector and Auditor (OIA) seek to assure effectiveness, efficiency and integrity in all NRC operations. In fiscal year 1985, OIA issued 15 audit reports, containing 148 recommendations, and 14 follow-up audit reports aimed at improving the operations of various NRC programs and activities. OIA also issued 41 investigative reports in response to allegations regarding the integrity of NRC operations and employees. In addition, 12 matters were referred to the Department of Justice for review and possible criminal prosecution.

Some of the reports issued during the report period are summarized below.

Review of NRC's Policies and Procedures Related to Differing Professional Opinions

In December 1984, OIA issued an audit report documenting the review of NRC's program for handling differing professional opinions. OIA found that the program was adequate. However, to achieve a more effective program, OIA recommended the establishment of a peer review group, an annual review of the program by the Special Review Panel, improvement in basic record-keeping, and closer compliance with administrative procedures.



NRC Commissioner Thomas M. Roberts was sworn in for a second term as a member of the Commission on July 12, 1985; his term will run to June 30, 1990. Commissioner Roberts has had first-hand experience in the manufacture of nuclear power plant components, having served as Chief Executive Officer and President of the Southern Boiler and Tank Works, Inc., of Memphis, Tenn., from 1969 to 1978. Mr. Roberts has also been an underwriting member of Lloyd's of London and a director of the Boyle Investment Company. He received his B.S. degree in Industrial Engineering from the Georgia Institute of Technology in 1959 and, subsequently, was commissioned an ensign in the U.S. Navy. He served as engineering officer during his three years aboard a Navy Destroyer.

Survey of the Office of The General Counsel

In a March 1985 audit report, OIA concluded that the Office of the General Counsel (OGC) was performing its mission as described in the NRC Manual and in the Code of Federal Regulations. However, OIA noted that management within OGC needed improvement; supervisory lines needed to be established; internal controls needed to be strengthened; and administrative procedures needed to be improved or established. OIA made recommendations to improve the internal management of OGC and to enhance OGC's service to the Commission.

Audit of Selected Aspects of NRC's Payroll Operations

In July 1985, OIA issued a report on the results of an audit of selected aspects of NRC's payroll operations. The report disclosed a number of areas in which OIA believed payroll operations needed improvement, including control procedures within payroll, use of payroll reports, training of payroll staff, and corrections of problems in the automated payroll system. OIA made recommendations to correct the problems noted.

Staff's Implementation of the Interim Procedures for Managing Plant-Specific Backfitting of Power Reactors

This June 1985 audit report to the Commission addressed the staff's implementation of the interim procedures to manage and control plant-specific backfits approved by the Commission in February 1984 as draft Manual Chapter 0514. The report documented an OIA conclusion that the interim procedures lacked the necessary specificity to allow the staff to adequately manage and control issued identified as backfits. In this regard, the procedures were directly solely to backfits identified by the staff and did not provide guidance on how to handle backfits identified by utilities. Consequently, many issues reported as backfits by utilities were tracked as backfits by NRR without a determination as to whether they were backfits. On May 1, 1985, the Deputy Director for Regional Operations and Generic Requirements implemented a revised draft Manual Chapter 0514 on an interim basis. Although this revision was a significant improvement over the earlier interim procedures, in OIA's opinion improvements were still needed. OIA made recommendations to improve the procedures contained in draft Manual Chapter 0514 and NRC's management of plant-specific backfits.

Review of Region V Management Of Diablo Canyon Allegations

As a result of a letter from an alleger to Commissioner Lando W. Zech, Jr., in which allegations of breach of confidentiality and the lack of feedback to allegers were made, OIA initiated a review of NRC allegations management with respect to the Diablo Canyon Nuclear Power Plant (Cal.). OIA interviewed allegers who had raised concerns about the NRC's handling of allegations about Diablo Canyon and evaluated 10 CFR 2.206 Petitions and other documents filed with NRC by allegers. As a result, OIA identified and reviewed more than 200 complaints regarding NRC's resolution of Diablo Canyon allegations. In July 1985, prior to issuance of the operating license for Diablo Canyon Unit 2, the Director, OIA, briefed the Commission on the results of the OIA review to date.

Review of NRC's Allegation Management System

This OIA review focused on NRC systems to account for, control, track and manage allegations. OIA looked at how NRC offices deal with allegations, how the allegations workload is coordinated within the agency, and how task forces are used to respond to allegations regarding Near Term Operating License applicants. OIA concluded that the NRC does have systems at place at NRC Headquarters and Regional Offices which provide an adequate framework for processing allegations. The OIA audit report contained 12 recommendations intended to correct specific problems identified during the review and to enhance the overall effectiveness of NRC's allegation management system.

Review of NRC Regionalization

In April 1985, OIA issued a report based on a review of NRC's regionalization program. The review focused on the regionalization of nuclear materials licensing, reactor operator licensing, and operating reactor licensing activities. OIA concluded that none of the programs was fully regionalized in terms of having all programs in place in the Regions and all guidance available to the regional staffs. The OIA report identified areas where management and oversight of the programs could be improved and made recommendations for improvement in the implementation of regionalized activities.

Inspection of the Office of Investigations

In accordance with the directive provided by the Chairman in his April 20, 1982 announcement of the formation of the Office of Investigations (OI), OIA performed an inspection of OI's program development, implementation and evaluation activities. The inspection included the activities of OI Headquarters and each of OI's 5 regional offices. OIA's May 10, 1985 report to the Commission contained 51 recommendations to improve the management, investigations and administration of OI.

Region II Actions Related to the Grand Gulf Nuclear Station

At the request of Congressman Edward J. Markey (D. Mass.), Chairman Palladino directed OIA to conduct an investigation of certain actions by Region II officials concerning the Grand Gulf Nuclear Station (Miss.). The OIA investigation disclosed that Region II officials inappropriately shared information with the licensee and allowed the nuclear plant to operate after problems with the training and qualifications of reactor operators at Grand Gulf had been identified. In October 1984, OIA issued a Report of Investigation to the Commission. A copy of the report was provided to Congressman Markey.

The NRC's Office of Inspection and Audit, at the behest of the NRC Chariman, undertook an investigation of the allegedly improper sharing, by NRC personnel, of information with the licensee for the Grand Gulf nuclear power plant (Miss.). The facility began commercial operations on July 1, 1985.

Reactor Operator Licensing Examination

OIA conducted an investigation into the circumstances surrounding the administration of an NRC Operator Licensing Examination at Indian Point Nuclear Power Plant Unit 3 (N.Y.). The OIA investigation disclosed that the Operator Licensing Examination administered at the plant by the NRC Licensing Examiner was invalid. The examination had been prepared by a training vendor who had no contract with the NRC and was concurrently employed by the Power Authority of the State of New York—licensee for the facility—to prepared the Indian Point Unit 3 reactor operator licensee candidates for the NRC examination. In October 1984, an OIA Report of Investigation was issued to the Executive Director for Operations.

CONTRACTING

Contracts with commercial firms for technical assistance, research work, and general purchases totaled approximately $52,500,000 in fiscal year 1985. Contracts under the Small Business Innovation Research Program totaled $1,500,000, and grants to educational and nonprofit institutions totaled $1,800,000.

PAPERWORK REDUCTIONS

Since 1981, the NRC has worked to improve the control of paperwork burdens imposed on nuclear plant licensees. The NRC's fiscal year 1981 base, as reported to the Office of Management and Budget under the Paperwork Reduction Act 1980, was 19.3 million hours of burden. The fiscal year 1984 base was 10.2 million hours, nearly half of the 1981 base; and in 1985, NRC achieved the 13 percent reduction called for in OMB's Information Collection Budget.

NRC has reduced the burden of reporting and record-keeping requirements through tightened management oversight, within the framework of a policy commitment to eliminate unnecessary regulatory burdens and TO allow licensees the flexibility to select the most cost-effective ways to satisfy NRC's safety objectives.

NRC LICENSE FEES

The NRC is authorized under Title V of the Independent Offices Appropriation Act of 1952 to collect fees for processing applications, permits, licenses and approvals and for routine and non-routine health and safety inspections. Fees billed in fiscal year 1985 totaled $86.2 million. All license and inspections fees are sent to the Department of Treasury for deposit as miscellaneous receipts. Table I shows a breakdown of these billings.

Table 1. FY 1985 License Fee Billings

Fees	*Materials*	*Facilities*	*Total*
Applications	$ 572,183	—	$ 572,183
Construction Permits	—	$ 1,701,893	1,701,893
Operating Licenses	—	71,375,591	71,375,591
Amendments	351,676	440,172	791,848
Renewals	411,432	—	411,432
Inspection Fees	1,274,458	9,889,705	11,164,163
Special Projects	102,642	64,916	167,558
	$2,712,391	$83,472,277	$86,184,668

Table 2. Cost of OL Issuances

FY 1985

Operating Licenses	*Issue Date*[1]	*Licensing Cost*[2]	*Inspection Cost*[2]	*Total Cost*	*Fee Billed*[3]
Limerick 1	10/26/84	$3,547,699	$708,893	$4,256,592	$3,077,400
Byron 1	10/31/84	1,953,244	804,161	2,757,405	2,757,405
Shoreham	12/07/84	3,074,102	1,098,248	4,172,350	3,077,400
Waterford	12/18/84	3,489,542	1,357,273	4,846,815	3,077,400
Palo Verde 1	12/31/84	1,812,337	737,957	2,550,294	2,550,294
Wolf Creek 1	3/11/85	1,439,403	633,652	2,073,055	2,073,055
Fermi 2	3/20/85	3,183,644	1,817,519	5,001,163	3,077,400
Diablo Canyon 2	4/26/85	1,574,310	452,912	2,027,222	2,027,222
River Bend 1	8/29/85	1,439,720	447,808	1,887,528	1,887,528

[1] The issue date is date of issuance of the initial operating license usually limited to 5 percent power operation.

[2] Costs shown represent those costs expended through the period ending 6/23/84 except for Waterford 3, Fermi 2, and Diablo Canyon 2 where the costs are given through the period ending 12/22/84.

[3] Fees are billed for the full cost of the review up to a maximum ceiling of $3,077,400 as established by regulation.

The total billings since fees were first imposed (October 1968 through September 1985) is $263.5 million. Of this amount, $6.5 million has been refunded to licensees because of a 1974 Supreme Court decision negating annual license fees. The refund program was completed on November 11, 1984, the date the statute of limitations expired.

The current schedule of fees, adopted June 20, 1984, provides that fees will be assessed for construction permits and operating licenses for power reactors every six months as the work progresses. Although many utilities were billed under the revised schedule for units undergoing licensing review, nine operating licenses were issued in fiscal year 1985. No construction permits were issued. Table II summarizes costs and billings for those reactors receiving an operating license.

PUBLIC COMMUNICATION

Public Information

Media Workshops. Four of the regional offices of the Nuclear Regulatory Commission conducted a series of one-day educational seminars for the fifth consecutive year for reporters and news editors from national wire services, broadcast networks, news magazines and daily newspapers. The seminars, which dealt with the fundamentals of nuclear power and the risks of exposure to radiation, were held in San Francisco, Cal., February 11; Glen Ellyn, Ill., March 21; Gainesville, Fla., March 26; Dallas, Tex., November 18; and Phoenix, Ariz., November 20.

Public Affairs. The NRC's Office of Public Affairs distributed press releases on Commission programs, rulemakings, public hearings, proposed fines against licensees, and other agency activities to the news media, scientific community, universities and the general public.

Headquarters Public Document Room

Persons interested in detailed information about commercial nuclear facilities have found the NRC's principal Public Document (PDR) a rich source of useful material. Located at 1717 H Street, N. W., in Washington, D. C., the PDR is a specialized documentation center that houses significant documents on nuclear regulation which have been made available to the public. Users of the center can have documents reproduced for a nominal fee.

Researchers in the PDR can examine copies of a wide variety of materials: NRC reports; transcripts and summaries of meetings; licenses and their amendments; existing and proposed regulations; and correspondence on technical, legal and administrative matters. Most of these documents are related specifically to nuclear power plants (their design, construction, operation and inspection) and to nuclear materials, including radioactive wastes (their use, transport and disposal). The PDR features extensive accession listing and an on-line computer data base.

The PDR contains about 1.4 million documents, and the collection is enlarged by an average of 318 new items every day. During an average month, the PDR serves 1200 users. The staff retrieves an average of 5,800 files per month containing multiple documents or microfiche for researchers on-site and provides about 1,800 documents in response to letters and telephone requests. The public purchased 3.1 million pages of documents and about 7,200 microfiche cards in fiscal year 1985.

Persons wishing to use or obtain additional information regarding the holdings, file organization, reference, reproduction services and procedures of the PDR may call (202) 634-3274 or write to the U. S. Nuclear Regulatory Commission, Public Document Room, Washington, D. C. 20555. A "Public Document Room Users' Guide" and "Public Document Room File Classification System" guide are available upon request. In addition, orientation sessions are provided for individuals or groups interested in using the facility, and training sessions are scheduled regularly for users in how to search the PDR automated bibliographic retrieval system (an on-line card catalogue).

Local Public Document Rooms

Through its local public document room (LPDR) program, the NRC makes document collections available to the public near the sites of proposed and operating nuclear power plants. These collections contain information about the licensing, construction, operation, inspection, and regulation of nearby nuclear facilities. They include documents dealing with health and safety, safeguards, environmental, and antitrust considerations. LPDR collections usually are located in university or public libraries that have copying facilities and are open to the public during the evening and on weekends. Currently, there are approximately 100 LPDRs in operation for nuclear power plants. (See Appendix 3 for a list of LPDR locations.)

To inform the public about the existence and availability of documents at the local level, NRC publishes a newsletter and conducts evening workshops at individual LPDR libraries. The workshops provide instruction to the public in identifying, locating, and retrieving information. A toll-free telephone number (1-800-638-8081) is available to library staffs and individuals who need rapid, convenient answers to questions about such topics as collection content, search strategies, use of reference tools and indices, and locating and retrieving information at LPDR sites.

H-Street Automation Activities

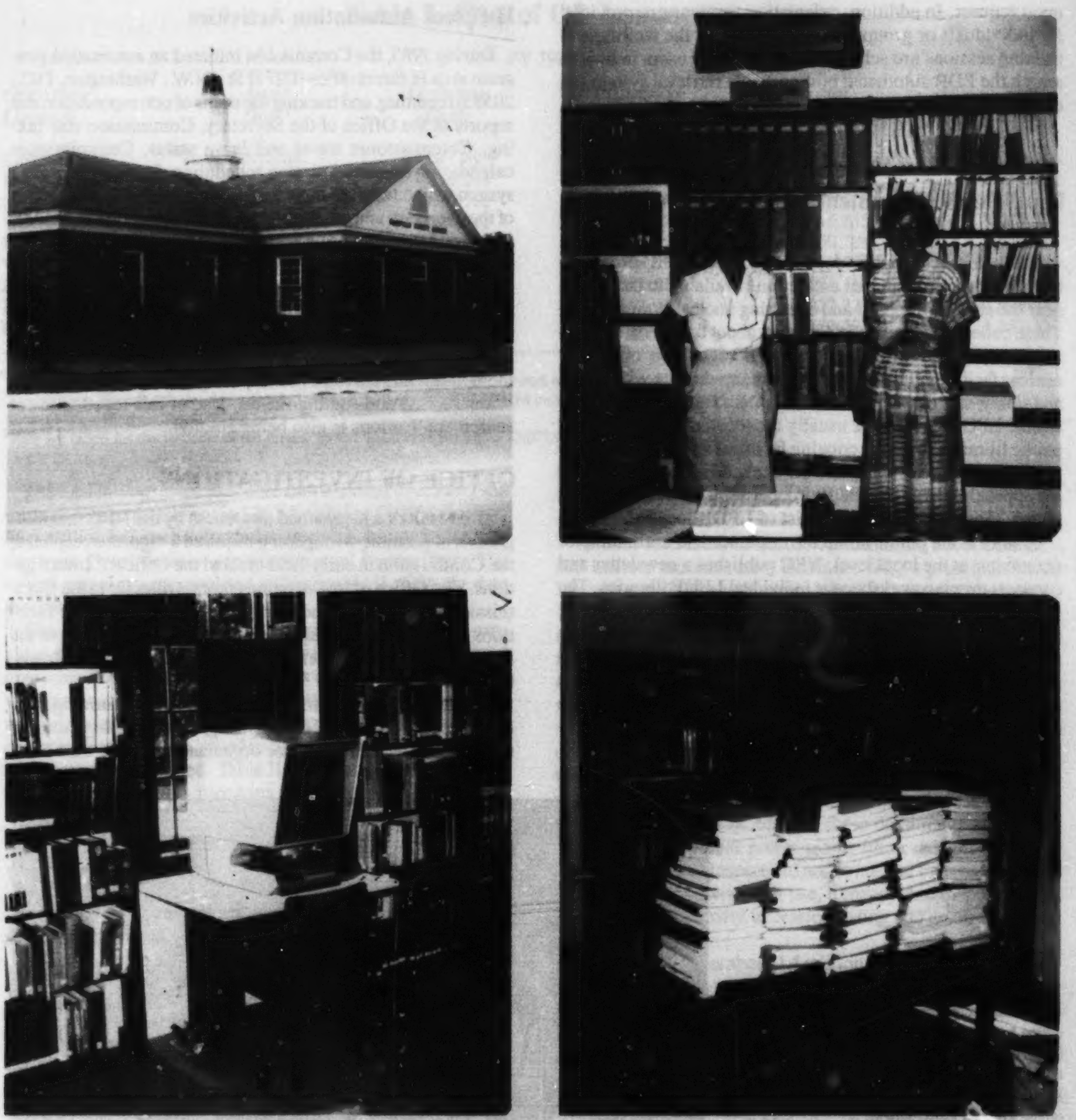
During 1985, the Commission initiated an automation program at its H-Street office (1717 H St., N.W., Washington, D.C., 20555) recording and tracking the status of correspondence and reports of the Office of the Secretary, Commission vote taking, Commissioner travel and leave status, Commissioner calendaring and Commission scheduling. It is an interactive system which receives input from Commissioners, the Office of the Secretary and other sources. This information is, in turn, combined according to a variety of informational criteria to produce planning, operating and historical data reports. In addition, a microcomputer-based system for the production and maintenance of service lists has been installed. Indexes of the Commission's adjudicatory dockets are now being produced. Still under development is the creation of a data base of memoranda and correspondence, designed to produce indexes, status reports and routing tickets. Use of full-text document search applications is also being studied.

OFFICE OF INVESTIGATIONS

As a result of a heightened awareness by the NRC that allegations of possible wrongdoing required a vigorous response, the Commission in early 1982 created the Office of Investigations. The Office of Investigations reports directly to the Commission. Supervision and direction of the investigative functions of the Office of Inspection and Enforcement and of the five Regional Offices were transferred to the Office of Investigations.

The Office of Investigations is charged with performing thorough, timely and objective investigations of allegations of wrongdoing by individuals or organizations other than NRC

Chairman Palladino met with more than 150 students from the Washington Workshops Foundation in 1985. The students, high-school student leaders from 26 States and several foreign countries, attended the one-week Congressional Seminar to learn first-hand about the workings of the Federal Government.



The NRC Local Public Document Room (LPDR) for the Virgil C. Summer nuclear power plant is located in the Fairfield County Library (above left) at nearby Winnsboro, S.C. Documents dealing with health and safety, safeguards, environmental and other considerations related to the nuclear plant are filed in a collection supervised by Sarah McMaster, Library Director, on the right, and Marge Brown, Library Assistant. Facilities for public review of the documents, such as the microfiche reader/printer (bottom left), are provided in the LPDR, and new materials are regularly added to the collection (bottom right). See Appendix 3 for a listing of all LPDRs and associated nuclear facilities.

employees or NRC contractors; this includes licensees, applicants, vendors or their contractors. Investigations are undertaken at the request of the Commission, the Executive Director for Operations, the Regional Administrators, or on the Office of Investigations own initiative. Some of the types of allegations within the Office of Investigations purview are charges of falsification of records, intimidation of Quality Control Inspectors, or deliberate violations of NRC regulations and requirements.

When an allegation is received, the staffs of the Office of Investigations and other NRC offices confer to determine the appropriate action office. An allegation, for instance, may be made of deliberate or willful wrongdoing, or it may focus on technical deficiencies in a plant. In some cases, a joint effort may be initiated by the Office of Investigations investigators and the Office of Inspection and Enforcement inspectors.

Upon completion of an investigation, a Report of investigation is issued within the Commission. Depending on the findings, this report may become the basis for an enforcement action by the Office of Inspection and Enforcement, the matter may be referred to the Department of Justice for prosecutory consideration, or the subject of investigation may be exonerated.

From its inception in July of 1982 to the end of fiscal year 1985, the Office of Investigations has opened 759 cases, and 581 of those have been closed. Thirty-six cases have been referred to the Department of Justice.

The Director of the Office of Investigations is Ben B. Hayes. The staff currently includes 35 professional investigators, some of whom have been recruited from the ranks of various Federal law enforcement agencies. The caseload of the office at any give time is about 175 cases.

COMMISSION HISTORY PROGRAM

The Commission History Program continues its study of the development of regulatory and licensing policies. An array of government records and collections of private manuscripts illuminating the period from 1963 to approximately 1970—during which nuclear regulation was the responsibility of the NRC's predecessor, the Atomic Energy Commission (AEC)—are being researched. That period, during which utilities ordered nuclear plants in large numbers, reactor size increased dramatically, the environmental movement gained momentum, and the AEC was involved in a series of controversies and regulatory issues, will be the focus of the second book in a multi-volumed historical series. The first volume, entitled *Controlling the Atom, The Beginnings of Nuclear Regulation, 1946-1962*, was published in December 1984 by the University of California Press. It is designed as a resource for the general reader as well as an authoritative reference for agency staff.

OFFICE OF SMALL AND DISADVANTAGED BUSINESS UTILIZATION/CIVIL RIGHTS

Small and Disadvantaged Business Utilization Program

The Small and Disadvantaged Business Utilization Program annually establishes procurement preference goals in response to provisions of Public Law 95-507, amending the Small Business Investment Act of 1957. During fiscal year 1985:

- It was estimated that $55 million in total prime contracts would be awarded in fiscal year 1985, and that the total amount of all prime contracts with individual dollar values over $10,000 would be $52 million. The actual total prime contracts and actual dollar awards over $10,000 were $55,338,709 and $51,969,068, respectively.
- It was estimated that small business prime awards with dollar values over $10,000 would be $17,400,000, or 33.46 percent. The actual achievement for small business prime awards with dollar values over $10,000 was $20,321,725, or 39.10 percent of the dollars reflected in Paragraph #1. This represents a percentage increase from FY-1984.
- Although NRC estimated that awards to 8(a) firms would be $6,600,000 or 12 percent in fiscal year 1985, awards to 8(a) firms were actually $6,127,372 or 11.07 percent of the total dollar amount of all prime contracts regardless of dollar value.
- The goal for prime contract awards having a value of $10,000 or more to small disadvantaged business firms other than 8(a) was $1,050,000 or 2.02 percent. The actual achievement was $929,569 or 1.79 percent of the dollars reported in Paragraph #1 using awards over $10,000 as the base.
- The estimate for prime contract awards to small business concerns owned and controlled by women was $920,000 or 1.66 percent. Awards to such firms were $1,311,940 or 2.37 percent of the total dollar amount of all prime contracts regardless of dollar value.
- The goal for subcontract awards to small business was $1,250,000 or 62.5 percent of total subcontracts awarded. Subcontracting achievement to small businesses was $2,052,047 or 68.94 percent of total subcontracts awarded. NRC's total subcontract dollar awards in fiscal year 1985 were $2,976,678. The goal was $2,000,000.
- The goal for subcontract awards to small disadvantaged businesses was $43,000 or 2.15 percent. Subcontracting awards to small disadvantaged businesses was $90,285 or 3.03 percent of total subcontract dollars awarded.

During the year, 93 interviews were conducted with firms wanting to do business with the NRC, and 47 follow-up meetings were arranged with NRC technical personnel.

The OSDBU/CR staff also participated in five major small business conferences, or "job fairs." Most noteworthy was the Annual Minority Enterprise Development Week (MED Week) observance. The events of this week recognized the numerous contributions of minority businesses to our national economy.

Civil Rights Program

The report year marked the implementation of the Agency's first consolidated Equal Employment Opportunity (EEO) program plan designed to promote affirmative action and to eliminate barriers to equal employment opportunity. The Office of Small and Disadvantaged Business Utilization/Civil Rights, in conjunction with the Office of Administration, took the lead in key program responsibilities such as data analysis, upward mobility programs, planning awareness and assistance programs, career counseling, complaint resolutions, and awards and recognition programs. Managers and supervisors remain responsible for areas such as EEO in promotions, training, hiring, awards/recognitions, and affirmative action to ensure greater representation of all groups of employees at all levels.

The EEO Consolidated Program Plan was fully implemented resulting in (1) the incorporation of EEO initiatives into the operating plans of Office Directors and Regional Administrators, (2) quarterly meetings with EEO Counselors and EEO Advisory Committees, (3) EEO seminars for Office Directors and Regional Administrators, and (4) the appointment of a Hispanic Employment Program Manager and a Black Affairs Coordinator.

On March 16, 1985, the Office of Small and Disadvantaged Business Utilization/Civil Rights briefed the Commission on the status of NRC's EEO/ Affirmative Action Plan goals, programs, and accomplishments. The resulting recommendation was to continue to strive for full and equal employment opportunity through program planning and the EEO Consolidiated Program Plan.

To ensure more realistic and achievable hiring goals, NRC contracted with Oak Ridge Associated Universities to determine the availability of women and minorities in the engineering and scientific professions in the civilian labor force. Data resulting from this study were used in preparing the annual update of the Multi-Year Affirmative Action Plan mandated by the Equal Employment Opportunity Commission.

Federal Women's Program

Fiscal year 1985 was marked by many successful initiatives to enhance equal employment opportunity for women. In the area of program development, a Federal Women's Program (FWP) brochure was developed and included in the Employee Orientation Packet for all new women employees; the Federal Women's Program Manager met with the Executive Director for Operations and his staff to discuss plans and objectives of the Federal Women's Program, to include the role managers must play to ensure program success. Regional Federal Women's Program Coordinators became members of the Regional EEO Committees and were encouraged to obtain training to enhance their effectiveness.

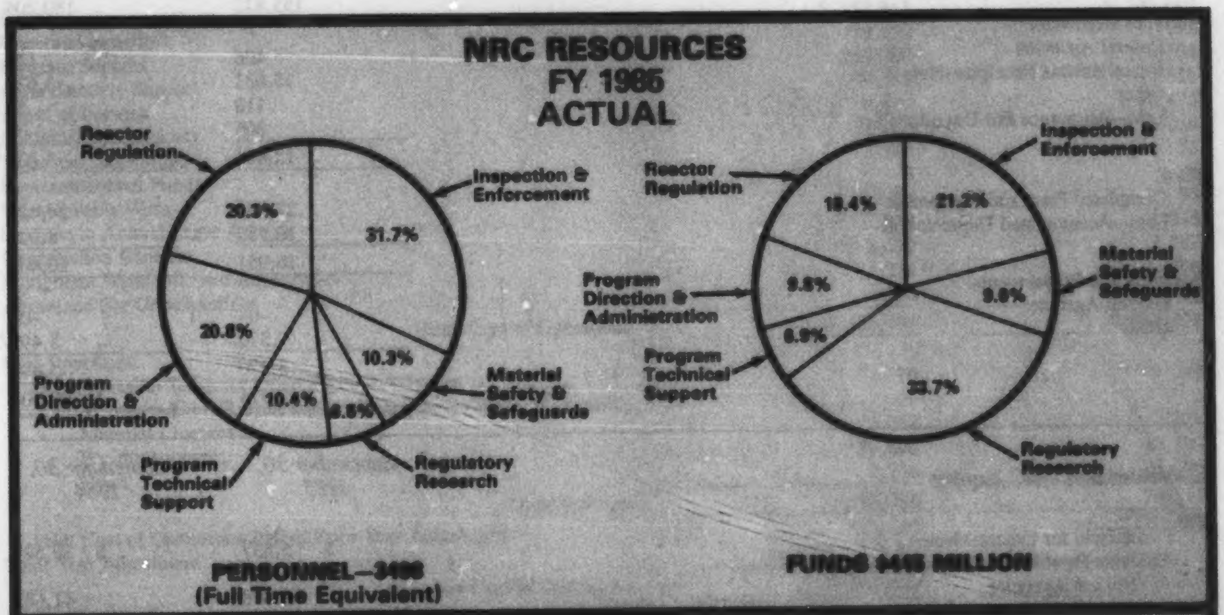
HYATT
REGENCY
BETHESDA

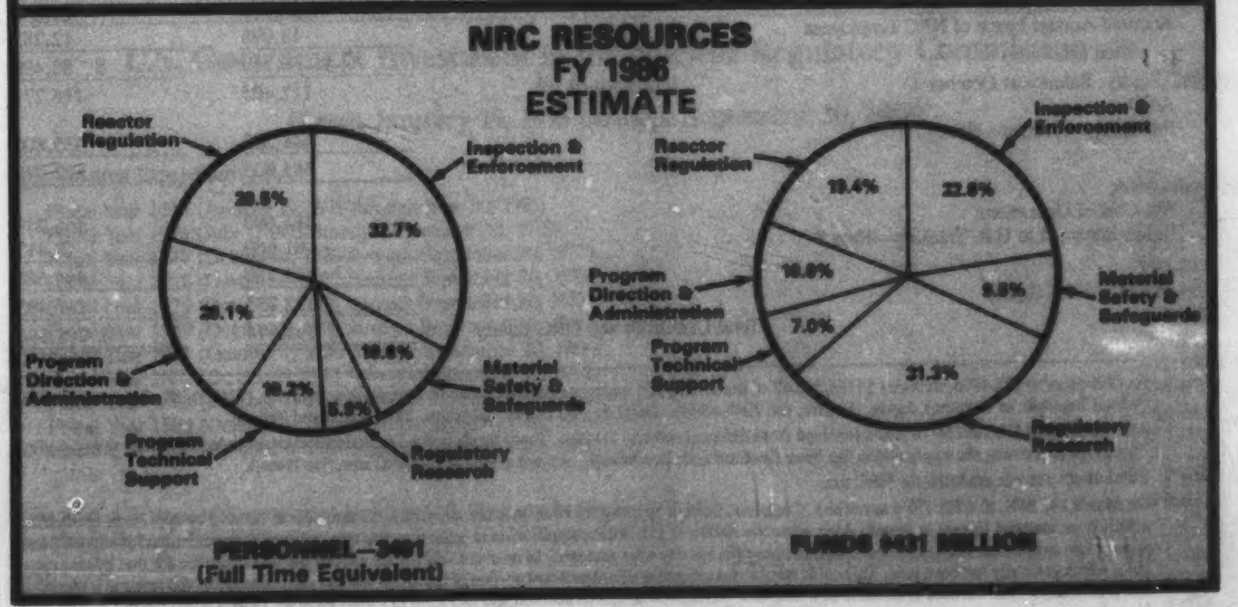
The NRC's observance of National Women's History Week included a special event sponsored by the NRC Federal Women's Program in Bethesda, Md., on March 4, 1985. The featured speaker for the occasion was Maureen Bunyan, television anchor for the CBS station in the Washington, D.C. area and a distinguished international news reporter. Ms. Bunyan was introduced by NRC Executive Director for Operations Victor Stello, Jr. NRC Chairman Nunzio J. Palladino also addressed the more than 500 persons attending the event.

Along with EEO initiatives, Federal Women's Program plans were incorporated into the operating plans of each Office Director and Regional Administrator. Exit interviews were conducted with women leaving NRC by the FWP Manager/Coordinators. And to improve the visibility of the FWP, a women's column was established in the NRC Newsletter, "News, Review and Comments," where seven articles for and about NRC women were published.

The Federal Women's Program Manager served as the Program Coordinator for the Agency's participation in the Office of Personnel Management (OPM) Women's Executive Leadership Program. NRC recommended five women, of whom OPM selected one.

There were other significant accomplishments in support of the goals of the Federal Women's Program. The Agency added a woman to its Senior Executive Service, and three women were selected to participate in the Senior Executive Service Candidate Program. Career counseling was provided women in Headquarters and in the five Regions. Several programs and activities concerning self-help and career development were sponsored by the Federal Women's Program and the Federal Women's Program Advisory Committee. In addition to participation in the Agency recruitment effort to employ more women, the Federal Women's Program Manager was also instrumental in the resolution of several informal EEO complaints.





FY 1984/1985 Financial Statements

Balance Sheet (in thousands)

Assets	*September 30, 1985*	*September 30, 1984*
Cash:		
Appropriated Funds in U.S. Treasury	$ 149,975	$ 169,677
Other—Notes 1 & 3	5,837	10,527
	155,812	180,204
Accounts Receivable:		
Federal Agencies	406	132
Miscellaneous Receipts—Note 2	15,631	3,623
Other	110	50
Less—Allowance For Uncollectibles	306	-0-
	15,841	3,805
Plant:		
Completed Plant and Equipment	27,534	24,429
Less—Accumulated Depreciation	10,583	8,026
	16,951	16,403
Advances and Prepayments:		
Federal Agencies	-0-	-0-
Other	3,370	5,492
	3,370	5,492
Total Assets	$ 191,974	$ 205,904

Liabilities and NRC Equity	*September 30, 1985*	*September 30, 1984*
Liabilities:		
Funds Held for Others—Notes 1 & 3	$ 5,837	$ 10,527
Accounts Payable and Accrued Expenses:		
Federal Agencies	24,058	41,683
Other	26,729	24,004
Accrued Annual Leave of NRC Employees	13,095	12,285
Total Liabilities	$ 69,719	$ 88,499
NRC Equity: Balance at October 1	117,405	116,271
Additions:		
Funds Appropriated—Net	448,200	465,800
	565,605	582,071
Deductions:		
Net Cost of Operations	361,690	446,249
Funds Returned to U.S. Treasury—Note 2	81,660	18,417
	443,350	464,666
Total NRC Equity	122,255	117,405
Total Liabilities and NRC Equity	$ 191,974	$ 205,904

Note 1. As of September 30, 1985, includes $5,014,555.48 of funds received under cooperative research agreements involving NRC, DOE, Euratom, France, Federal Republic of Germany, Japan, Austria, the Netherlands, Belgium, the United Kingdom, Italy, Korea and Taiwan.

Also included is $720,947.90 of funds received from deferred revenue billings. These funds will be refunded and/or recorded as earned revenue after the cost of processing the applications has been finalized and, accordingly, are not available for NRC use. See Note 3.

Note 2. These funds are not available for NRC use.

Note 3. On March 24, 1978, 10 CFR 170 was revised. Contained therein by category of license are maximum fee amounts to be paid by applicants at the time a facility or material license is issued. Also, after the review of the license application is complete, the expenditures for professional manpower and appropriate support services are to be determined and the resultant fee assessed. In no event will the fee exceed the maximum fee for that license category, which generally has been paid. This could involve the refunding of a significant portion of the initial amount paid. Therefore, the revenue is recorded in a Deferred revenue account at the time of billing and is removed from this account and recorded in Funds Held for Others when the bill is paid. See Note 1.

FY 1984/1985 NRC Statement of Operations
(in thousands)

	Fiscal Year, 1985 (October 1, 1984, thru September 30, 1985)	*Fiscal Year, 1985 (October 1, 1983, thru September 30, 1984)*
Personnel Compensation	$ 152,943	$ 143,643
Personnel Benefits	19,150	16,196
Program Support	219,700	250,608
Administrative Support	49,558	41,516
Travel of Persons	8,887	10,475
Equipment (Technical)	295	359
Taxes and Indemnities	749	72
Representational Funds	3	2
Reimbursable Work	76	90
Increase in Annual Leave Accrual	810	1,013
Depreciation Expense	2,679	2,287
Equipment Write-Offs and Adjustments	278	200
Allowance For Uncollectibles	306	-0-
Total Cost of Operations	$ 455,434	$ 466,461
Less Revenues:		
Reimbursable Work for Other Federal Agencies	76	92
Fees (Deposited in U.S. Treasury as Miscellaneous Receipts—Note 2):		
Material Licenses	3,062	2,274
Facility Licenses	87,890	14,750
Other	2,716	3,096
Total Revenue	93,744	20,212
Net Cost of Operations Before Prior Year Adjustments	361,690	446,249
Prior Year Adjustment	-0-	-0-
Net Cost of Operations	$ 361,690	$ 446,249

U.S. Government Investment in the Nuclear Regulatory Commission
(in thousands)

(From January 19, 1975 through September 30, 1985)

Appropriation Expenditures:		
Fiscal Year 1975 (January 19, 1975 through June 30, 1975)	$ 52,792	
Fiscal Year 1976 (July 1, 1975 through September 30, 1976)	226,248	
Fiscal Year 1977 (October 1, 1976 through September 30, 1977)	230,559	
Fiscal Year 1978 (October 1, 1977 through September 30, 1978)	270,877	
Fiscal Year 1979 (October 1, 1978 through September 30, 1979	309,493	
Fiscal Year 1980 (October 1, 1979 through September 30, 1980)	377,889	
Fiscal Year 1981 (October 1, 1980 through September 30, 1981)	416,867	
Fiscal Year 1982 (October 1, 1981 through September 30, 1982)	441,902	
Fiscal Year 1983 (October 1, 1982 through September 30, 1983)	514,613	
Fiscal Year 1984 (October 1, 1983 through September 30, 1984)	462,084	
Fiscal Year 1985 (October 1, 1984 through September 30, 1985)	467,902	
		$ 3,771,226
Unexpended Balance of Appropriated Funds in U.S. Treasury September 30, 1985		149,975
Transfer of Refunds Receivable from Atomic Energy Commission, January 19, 1975		429
Funds Appropriated—Net		3,921,630
Less:		
Funds Returned to U.S. Treasury—Note 2		222,372
Assets and Liabilities Transferred from Other Federal Agencies Without Reimbursement		1,673
Net Cost of Operations from January 19, 1975 through September 30, 1985		3,575,330
Total Deductions		3,799,375
NRC Equity at September 30, 1985 as Shown on Balance Sheet		$ 122,255

Appendix 1

NRC ORGANIZATION

(As of December 31, 1985)

COMMISSIONERS

Nunzio J. Palladino, Chairman
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal
Lando W. Zech, Jr.

The Commission Staff

General Counsel, Herzel H.E. Plaine
Office of Policy Evaluation, John E. Zerbe, Director
Office of Public Affairs, Joseph J. Fouchard, Director
Office of Congressional Affairs, Carlton C. Kammerer, Director
Office of Inspector and Auditor, Sharon R. Connelly, Director
Secretary of the Commission, Samuel J. Chilk
Office of Investigations, Ben B. Hayes, Director

Advisory Committee and Panels

Advisory Committee on Reactor Safeguards, David A. Ward, Chairman
Atomic Safety & Licensing Board Panel, B. Paul Cotter, Jr., Chairman
Atomic Safety & Licensing Appeal Panel, Alan S. Rosenthal, Chairman

EXECUTIVE DIRECTOR FOR OPERATIONS

Executive Director for Operations, William J. Dircks
Deputy Executive Director for Operations, Jack W. Roe
Deputy Executive Director for Regional Operations and Generic Requirements, Victor Stello, Jr.
Assistant for Operations, Thomas A. Rehm

Program Offices

Office of Nuclear Reactor Regulation, Harold R. Denton, Director
Office of Nuclear Material Safety and Safeguards, John G. Davis, Director
Office of Nuclear Regulatory Research, Robert B. Minogue, Director
Office of Inspection and Enforcement, James M. Taylor, Director

Staff Offices

Office of Administration, Patricia G. Norry, Director
Executive Legal Director, Guy H. Cunningham
Office of Resource Management/Controller, Ronald M. Scroggins
Office of International Programs, James R. Shea, Director
Office of State Programs, G. Wayne Kerr, Director
Office of Analysis and Evaluation of Operational Data, Clemens J. Heltemes, Jr., Director
Office of Small and Disadvantaged Business Utilization/ Civil Rights, William B. Kerr, Director

Regional Offices

Region I Philadelphia, Pa., Thomas E. Murley, Regional Administrator
Region II Atlanta, Ga., James P. O'Reilly, Jr., Regional Administrator
Region III Chicago, Ill., James G. Keppler, Regional Administrator
Region IV Dallas, Tex., Robert D. Martin, Regional Administrator
Region V San Francisco, Cal., John B. Martin, Regional Administrator

The NRC is responsible for licensing and regulating nuclear facilities and materials and for conducting research in support of the licensing and regulatory process, as mandated by the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and the Nuclear Nonproliferation Act of 1978; and in accordance with the National Environmental Policy Act of 1969, as amended, and other applicable statutes. These responsibilities include protecting public health and safety, protecting the environment, protecting and safeguarding materials and plants in the interest of national security, and assuring conformity with antitrust laws. Agency functions are performed through: standards-setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits and licenses; inspection, investigation and enforcement; evaluation of operating experience; and regulatory research. The Commission itself is composed of five members, appointed by the President and confirmed by the Senate, one of whom is designated by the President as Chairman. The Chairman is the principal executive officer and the official spokesman of the Commission.

The Executive Director for Operations directs and coordinates the Commission's operational and administrative activities among the program and support staff offices described below and also coordinates the development of policy options for Commission consideration. The EDO reports directly to the Chairman.

The Office of Nuclear Reactor Regulation licenses nuclear power, test and research reactors under a two-phase process. A construction permit is granted before facility construction can begin and an operating license is issued before fuel can be loaded. NRR reviews license applications to assure that each proposed facility can be built and operated without undue risk to the health and safety of the public and with minimal impact on the environment. NRR monitors operating reactor facilities during their lifetime through decommissioning.

The Office of Nuclear Material Safety and Safeguards is responsible for the licensing and regulation of facilities and materials associated with the processing, transport and handling of nuclear materials, and with the disposal of nuclear waste; the office also regulates uranium recovery facilities. NMSS reviews and assesses safeguards against potential threats, thefts and sabotage for licensed facilities, including reactors, working closely with other NRC offices in coordinating safety and safeguards programs and in recommending research, standards and policy options necessary for their successful operation.

The Office of Nuclear Regulatory Research plans and conducts a comprehensive research and standards program that is deemed necessary for the performance of the Commission's licensing and regulatory functions and that is responsive to current and future NRC needs. The program covers such areas as facility operation, engineering technology, accident evaluation, probabilistic risk analysis, siting, health, and waste management.

The Office of Inspection and Enforcement develops and oversees programs of inspection of nuclear facilities and materials licensees in order to determine whether those facilities are constructed and operations are conducted in compliance with license provisions and Commission regulations; in order to identify conditions that may adversely affect the protection of the public health and safety, of nuclear materials and facilities, or of the environment; and in order to provide a basis for recommending issuance or denial of licenses. IE develops and oversees a program of investigation of accidents, incidents and allegations of improper actions that involve nuclear material and facilities; enforces NRC regulations and license provisions; and manages and directs all NRC actions related to emergency preparedness, including evaluations of State and local emergency plans performed by the Federal Emergency Management Agency (FEMA). It also performs audits of its programs as carried out by NRC Regional Offices.

The Regional Offices are under the supervision and direction of the Executive Director for Operations and carry out NRC regulatory programs originating in various Headquarters Offices.

THE COMMISSION STAFF

The Office of the Secretary provides general management services to support the Commission and to implement Commission decisions, advises and assists the Commission and staff on the planning, scheduling and conduct of Commission business; prepares for and records Commission meetings; manages the Commission staff paper system and monitors the status of all items requiring action; integrates automated data processing and office automation initiatives into the Commission's administrative system, maintains a forecast of matters for future Commission consideration; processes and controls Commission correspondence; maintains the Commission's official records; maintains the official Commission adjudicatory and rulemaking dockets and serves Commission issuances in all adjudicatory matters and public proceedings; administers the NRC Historical Program; and directs and administers the NRC Public Document Room.

The Office of the General Counsel serves the Commission in a variety of legal capacities. The Office assists the Commission in the review of Appeal Board decisions, of petitions seeking direct Commission relief, and of rulemaking proceedings; the Office drafts the legal documents necessary to carry out the Commission's decisions. The General Counsel provides a legal analysis of proposed legislation affecting the Commission's functions and assists in drafting legislation and preparing testimony. The General Counsel also represents the Commission in court proceedings, frequently in conjunction with the Department of Justice.

The Office of Policy Evaluation plans and manages activities involved in performance of an independent review of judgments and positions developed by the NRC staff which require policy determinations by the Commission. The Office also conducts analyses and projects which are either self-generated or requested by the Commission.

The Office of Investigations conducts, supervises and assures quality control of investigations of licensees, applicants, contractors or vendors, including the investigation of all allegations of wrongdoing by other than NRC employees and contractors. The Office develops policy, procedures and standards for these activities.

The Office of Inspector and Auditor investigates to ascertain the integrity of all NRC operations; investigates allegations of NRC employee misconduct, equal employment and civil rights complaints, and claims for personal property loss or damage; conducts the NRC's internal audit activities; and hears individual employee concerns regarding Commission activities, under the Agency's "open door" policy. The Office develops policies governing the Commission's financial and management audit program and is the agency contact with the General Accounting Office on this function. The Office refers criminal matters to the Department of Justice and maintains liaison with law enforcement agencies.

The Office of Public Affairs plans and administers NRC's program to inform the public of Commission policies, programs and activities and keeps NRC management informed of public affairs activities of interest to the Commission. OPA reports directly to the Chairman.

The Office of Congressional Affairs provides advice and assistance to the Commission and senior staff on congressional matters, coordinates NRC's congressional relations activities, and maintains liaison for the Commission with congressional committees and members of Congress. OCA reports directly to the Chairman.

SUPPORT STAFF

The Office of Administration directs the Agency's programs for organization and personnel management; security and classification; technical information and document control; facilities and materials license fees; contracting and procurement; rules, proceedings and document services, including administration of Freedom of Information Act and Privacy Act requests; management development and training; telecommunications; transportation services; management of space; and other administrative services.

The Office of Resource Management develops and maintains NRC's financial and manpower management programs, including policies, procedures and standards of accounting, budgeting, cost analysis, resource planning and analysis, and automated data processing systems development and support. The Office provides management information for other offices and issues special reports from the NRC to Congress, other government agencies and the public. The Office assists NRC offices in statistical matters and in the budget process, keeping the EDO and Commission informed on programs and issues of significance. RM also maintains liaison with the Office of Management and Budget, the Congress, other agencies of government, and the private sector.

The Office of the Executive Legal Director provides legal advice and services to the Executive Director for Operations and staff, including representation in administrative proceedings involving the licensing of nuclear facilities and materials, and the enforcement of license conditions and regulations; counseling with respect to safeguards matters, contracts, security, patents, administration, research, personnel, and the development of regulations to implement applicable Federal statutes.

The Office of International Programs plans and implements programs of international nuclear safety cooperation, creating and maintaining relationships with foreign regulatory agencies and international organizations; coordinates NRC export-import and international safeguards policies; issues export and import licenses; and coordinates responses by NRC to other agencies related to export-import actions and issues.

The Office of State Programs directs programs related to regulatory relationships with State governments and organizations and interstate bodies, manages the NRC State Agreements program, administers the indemnification program and performs financial qualifications reviews of applicants and licensees. The Office also verifies that applicants are not in violation of the antitrust laws.

The Office for Analysis and Evaluation of Operational Data provides agency coordination for the collection, storage, and retrieval of operational data associated with licensed activities, analyzes and evaluates such operational experience and feeds back the lessons of that experience to NRC licensing, standards and inspections activities. The Office oversees action taken in response to the feedback and assesses the overall effectiveness of the agency-wide operational safety data program, serving as a focal point for interaction with the ACRS and industry groups involved in operational safety data analysis and evaluation.

The Office of Small and Disadvantaged Business Utilization/Civil Rights develops and implements the NRC's program in accordance with the Small Business Act, as amended, insuring that appropriate consideration is given to labor surplus area firms and women-owned businesses. The Office develops and recommends NRC policy providing for equal employment opportunity and develops, monitors and evaluates the affirmative action program to assure compliance with the policy. The Office also serves as contact with local and national public and private organizations with related interests.

OTHER OFFICES

The Advisory Committee on Reactor Safeguards is a statutory committee of 15 scientists and engineers advising the Commission on safety aspects of proposed and existing nuclear facilities and on the adequacy of proposed reactor safety standards and performing such other duties as the Commission may request. The Committee conducts a continuing study of reactor safety research and submits an annual report to the Congress. The Committee also administers the ACRS Fellowship Program.

The Atomic Safety and Licensing Board Panel is a panel of lawyers and others with expertise in various technical fields from which three-member Licensing Boards are drawn to conduct public hearings and make such intermediate or final decisions as the Commission may authorize in proceedings to grant, amend, suspend or revoke NRC licenses.

The Atomic Safety and Licensing Appeal Panel is a panel from which three-member Appeal Boards are selected to exercise the authority and perform the review functions which would otherwise be carried out by the Commission in certain licensing proceedings. Licensing Board decisions are reviewable by an Appeal Board, either in response to an appeal or on its own initiative. The Appeal Board's decision is also subject to review by the Commission in response to an appeal for discretionary review or on its own initiative.

Appendix 2

NRC Committees and Boards

Advisory Committee on Reactor Safeguards

The Advisory Committee on Reactor Safeguards (ACRS) is a statutory committee established to advise the Commission on the safety aspects of proposed and existing nuclear facilities and the adequacy of proposed reactor safety standards, and to perform such other duties as the Commission may request. As of December 31, 1985, the members were:

CHAIRMAN: MR. DAVID A. WARD, Chairman, Research Manager, Reactor Safety Research, E. I. du Pont de Nemours & Company, Savannah River Laboratory, Aiken, S.C.

VICE-CHAIRMAN: DR. HAROLD W. LEWIS, Professor of Physics, Department of Physics, University of California, Santa Barbara, Cal.

DR. MAX W. CARBON, Professor and Chairman of Nuclear Engineering Department, University of Wisconsin, Madison, Wisc.

MR. JESSE C. EBERSOLE, retired Head Nuclear Engineer, Division of Engineering Design, Tennessee Valley Authority, Knoxville, Tenn.

DR. WILLIAM KERR, Professor of Nuclear Engineering and Director of the Office of Energy Research, University of Michigan, Ann Arbor, Mich.

DR. CARSON MARK, Retired Division Leader, Los Alamos Scientific Laboratory, Los Alamos, N.M.

MR. CARLYLE MICHELSON, retired Principal Nuclear Engineer, Tennessee Valley Authority and Retired Director, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C.

DR. DADE W. MOELLER, Professor of Engineering in Environmental Health and Director, Office of Continuing Education, School of Public Health, Harvard University, Boston, Mass.

DR. DAVID OKRENT, Chairman, School of Engineering and Applied Science, University of California, Los Angeles, Cal.

MR. GLENN A. REED, Retired Plant Manager, Pt. Beach Nuclear Power Plant, Wisconsin Electric Power Company, Two Rivers, Wisc.

DR. FORREST J. REMICK, Acting Vice President for Research and Graduate Studies and Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.

DR. PAUL G. SHEWMON, Professor and Chairman of Metallurgical Engineering Department, Ohio State University, Columbus, Ohio

DR. CHESTER P. SIESS, Professor Emeritus of Civil Engineering, University of Illinois, Urbana, Ill.

MR. CHARLES J. WYLIE, Retired Chief Engineer, Electrical Division, Duke Power Company, Charlotte, N.C.

Atomic Safety and Licensing Board Panel

PANEL MEMBERS:

CHIEF ADMINISTRATIVE JUDGE B. PAUL COTTER, JR., ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Executive) ROBERT M. LAZO, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Technical) FREDERICK J. SHON, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE GEORGE C. ANDERSON, Marine Biologist, University of Washington, Seattle, Wash.

JUDGE CHARLES BECHHOEFER, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE LAWRENCE BRENNER, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE GLENN O. BRIGHT, ASLBP Engineer, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE A. DIXON CALLIHAN, Retired Physicist, Union Carbide Corporation, Oak Ridge, Tenn.

JUDGE JAMES H. CARPENTER, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE HUGH K. CLARK, Retired Attorney, E.I. duPont deNemours & Company, Kennedyville, Md.

JUDGE RICHARD F. COLE, ASLBP Environmental Scientist U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE FREDERICK P. COWAN, Retired Physicist, Brookhaven National Laboratory, Boca Raton, Fla.

JUDGE MICHAEL A. DUGGAN, Economist, University of Texas, Austin, Tex.

JUDGE GEORGE A. FERGUSON, Physicist, Howard University, Washington, D.C.

JUDGE HARRY FOREMAN, Medical Doctor, University of Minnesota, Minneapolis, Minn.

JUDGE RICHARD F. FOSTER, Environmental Scientist, Sunriver, Ore.

JUDGE JOHN H. FRYE, III, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE JAMES P. GLEASON, Attorney, Silver Spring, Md.

JUDGE ANDREW C. GOODHOPE, Retired Administrative Law Judge, Federal Trade Commission, Wheaton, Md.

JUDGE CADET H. HAND, JR., Marine Biologist, University of California, Bodega Bay, Cal.

JUDGE JERRY HARBOUR, ASLBP Environmental Scientist U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE DAVID L. HETRICK, Nuclear Engineer, Lawrence Livermore Laboratory, Livermore, Cal.
JUDGE ERNEST E. HILL, Nuclear Engineer, Lawrence Livermore Laboratory, Livermore, Cal.
JUDGE FRANK F. HOOPER, Marine Biologist, University of Michigan, Ann Arbor, Mich.
JUDGE HELEN F. HOYT, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE ELIZABETH B. JOHNSON, Nuclear Engineer, Oak Ridge National Laboratory, Oak Ridge, Tenn.
JUDGE WALTER H. JORDAN, Retired Physicist, Oak Ridge Laboratories, Oak Ridge, Tenn.
JUDGE JAMES L. KELLEY, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE JERRY R. KLINE, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE JAMES C. LAMB, III, Sanitary Engineer, University of North Carolina, Chapel Hill, N.C.
JUDGE GUSTAVE A. LINENBERGER, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE LINDA W. LITTLE, Environmental Biologist, L.W. Little Associates, Raleigh, N.C. JUDGE EMMETH A. LUEBKE, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE MORTON B. MARGULIES, ASLBP Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE KENNETH A McCOLLOM, Electrical Engineer, Oklahoma State University, Stillwater, Okla.
JUDGE GARY L. MILHOLLIN, Attorney, Catholic University of America, Washington, D.C.
JUDGE MARSHALL E. MILLER, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE PETER A. MORRIS, ASLBP, Physicist, US Nuclear Regulatory Commission, Bethesda, Md.
JUDGE OSCAR H. PARIS, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission Bethesda, Md.
JUDGE PAUL W. PURDOM, Retired Environmental Engineer, Decatur, Ga.
JUDGE DAVID R. SCHINK, Oceanographer, Texas A&M University College Station, Tex.
JUDGE IVAN W. SMITH, ASLBP Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE MARTIN J. STEINDLER, Chemist, Argonne National Laboratory, Argonne, Ill.
JUDGE QUENTIN J. STOBER, Biologist, University of Washington, Seattle, Wash.
JUDGE SEYMOUR WENNER, Retired Administrative Law Judge, Postal Rate Commission, Chevy Chase, Md.
JUDGE SHELDON J. WOLFE, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

PROFESSIONAL STAFF:

DONNA D. DUER, Legal Intern, U.S. Nuclear Regulatory Commission, Bethesda, Md.
CHARLES J. FITTI, Executive Secretary, U.S. Nuclear Regulatory Commission, Bethesda, Md.
ELLEN C. GINSBERG, Legal Intern; U.S. Nuclear Regulatory Commission, Bethesda, Md.
ELVA W. LEINS, Assistant Executive Secretary, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DAVID L. PRESTEMON, Legal Counsel to the Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JACK G. WHETSTINE, Hearing Support Supervisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.

Atomic Safety and Licensing Appeal Panel

An Atomic Safety and Licensing Appeal Board, established September 18, 1969, was delegated the authority to perform the review function which would otherwise be performed by the Commission in proceedings on applications for licenses or authorizations in which the Commission had a direct financial interest, and in such other licensing proceedings as the Commission might specify.

In view of the increase in the number of proceedings subject to administrative appellate review, the Atomic Safety and Licensing Appeal Panel was established on October 25, 1972, from whose membership three-member panel Appeal Boards could be designated for each proceeding in which the Commission had delegated its authority to an appeal board. At the same time, the Commission modified its rules to delegate authority to Appeal Boards in all proceedings involving the licensing of production and utilization facilities (for example, power reactors).

Pursuant to subsection 201 (g)(l) of the Energy Reorganization Act of 1974, the functions performed by appeal boards were specifically transferred to the Nuclear Regulatory Commission. The Commission appoints members to the Appeal Panel, and the Chairman of the panel designates a three-member appeal board for each proceeding. The Commission retains review authority over decisions and actions of Appeal Boards. The Appeal Board Panel, on October 1, 1985, was composed of the following persons:

FULL-TIME MEMBERS:

ALAN S ROSENTHAL, Appeal Panel Chairman, U.S. Nuclear Regulatory Commission, Bethesda, Md.
GARY J. EDLES, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DR. REGINALD L. GOTCHY, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
CHRISTINE N. KOHL, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
THOMAS S. MOORE, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
HOWARD A WILBER, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
PART-TIME MEMBER:
DR. W. REED JOHNSON, Professor of Nuclear Engineering, University of Virginia, Charlottesville, Va.

PROFESSIONAL STAFF:

JOHN CHO, Counsel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
THOMAS G. SCARBROUGH, Technical Advisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DAVID P. GESSER, Law Clerk, U.S. Nuclear Regulatory Commission, Bethesda, Md.

Advisory Committee on Medical Uses of Isotopes

The Advisory Committee on Medical Uses of Isotopes (ACMUI) was established in July 1958. The ACMUI, composed of qualified physicians and scientists, considers medical questions referred to it by the NRC staff and renders expert opinions regarding the medical uses of radioisotopes. The ACMUI also advises the NRC staff, as required, on matters of policy. Members are employed under yearly personal services contracts. As of September 30, 1985, the members were:

RICHARD E. CUNNINGHAM, Chairman, ACMUI, Director, Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Silver Spring, Md.

DR. VINCENT P. COLLINS, Medical Director, Houston Institute for Cancer Research, Diagnosis and Treatment, Houston, Tex.

DR. SALLY J. DE NARDO, Director, Nuclear Hematology-Oncology, Department of Nuclear Medicine, University of California Davis Medical Center, Sacramento, Cal.

DR. JACK K. GOODRICH, Radiology Associates of Erie, Erie, Pa.

DR. MELVIN L. GRIEM, Professor and Director, Chicago Tumor Institute, University of Chicago, Chicago, Ill.

DR. NILO E. HERRERA, Director, Department of Laboratory Medicine, Danbury Hospital, Danbury, Conn.

DR. B. LEONARD HOLMAN, Chief, Clinical Nuclear Medicine, Department of Radiology, Brigham and Women's Hospital Boston, Mass.

DR. GERALD M. POHOST, Director, Division of Cardiovascular Disease, University of Alabama at Birmingham, Birmingham, Ala.

DR. EDWARD W. WEBSTER, Director, Department of Radiation Physics, Massachusetts General Hospital, Boston, Mass.

DR. DAVID H. WOODBURY, Director, Nuclear Medicine Section, Wayne County General Hospital, Westland, Mich.

Advisory Panel for the Decontamination of Three Mile Island Unit 2

The Advisory Committee for the Decontamination of Three Mile Island, Unit 2, was established in October 1980. Its purpose is to obtain input and views from the residents of the Three Mile Island area and afford Pennsylvania government officials an opportunity to participate in the Commission's decisional process regarding cleanup plans for Three Mile Island, Unit 2. The Panel consists of the following members representing agencies of the Commonwealth of Pennsylvania, local government authorities in the vicinity of the Three Mile Island facility, the scientific community and persons having their principal place of residence in the vicinity of the facility.

ARTHUR E. MORRIS, Chairman, Mayor of Lancaster, Pa.

JOSEPH J. DINNUNO, Private Consultant, Annapolis, Md.

THOMAS B. COCHRAN, Senior Staff Scientist, Natural Resources Defense Council, Washington, D. C.

THOMAS GERUSKY, Director of the Pennsylvania Bureau of Radiation Protection, Department of Environmental Resources, Harrisburg, Pa.

ELIZABETH MARSHALL, resident of York, Pa.

KENNETH L. MILLER, Director of the Division of Health Physics and Associate Professor of Radiology, Milton S. Hershey Medical Center, Hershey, Pa.

GORDON ROBINSON, Associate Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.

JOEL ROTH, resident of Elizabethville, Pa.

THOMAS SMITHGALL, resident of Lancaster, Pa.

ANN TRUNK, resident of Middletown, Pa.

NEIL WALD, Professor of Radiation Health, Department of Radiology, University of Pittsburgh, Pittsburgh, Pa.

Appendix 3

Local Public Document Rooms

Copies of most documents originating in the NRC or submitted to it for review are placed in the Commission's Public Document Room (PDR) at 1717 H Street, N.W., Washington, D.C., for public inspection. Other PDRs on NRC premises include the rooms at the Willste Building, 7915 Eastern Avenue, Silver Spring, Md., and in the five Regional Offices (the latter for documents related to nuclear material licenses, i.e., most byproduct and source material licenses). In addition, documents related to licensing proceedings or licensed operation of specific facilities are made available in local PDRs established in the vicinity of each proposed or existing nuclear facility. The locations of the local PDRs, the names of the persons to contact, and the names of the facilities for which documents are retained are listed below. (N.B. Updated listings of local PDRs may be obtained by writing to the Local Public Document Room Branch, Division of Rules and Records, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.)

ALABAMA

- Mrs. Maude S. Miller Head
  Librarian Athens Public Library
  South Street
  Athens, Ala. 35611
  Browns Ferry Nuclear Power Station

- Ms. Bettye Forbus
  Director Houston-Love Memorial Library 212 W. Burdeshaw Street
  P.O. Box 1369
  Dothan, Ala. 36302
  Joseph M. Farley Nuclear Plant

- Mrs. Peggy McCutchen
  Director Scottsboro Public Library
  1002 S. Broad Street
  Scottsboro, Ala. 35768
  Bellefonte Nuclear Plant

ARIZONA

- Ms. Stefanie Moritz
  Documents Librarian
  Sciences Phoenix Public Library
  12 East McDowell Road
  Phoenix, Ariz. 85004
  Palo Verde Nuclear Generating Station

ARKANSAS

- Mrs. Miriam W. Cook
  Documents Librarian
  Tomlinson Library
  Arkansas Tech. University
  Russellville, Ark. 72801
  Arkansas Nuclear One

CALIFORNIA

- Ms. Judy Klapproth
  Director
  Eureka-Humboldt County Library
  636 F Street
  Eureka, Cal. 95501
  Humboldt Bay Power Plant

- Mrs. Fontayne Holmes
  Senior Librarian
  WESt Los Angeles Regional Library
  11360 Santa Monica Boulevard
  Los Angeles, Cal. 90025
  UCLA Training Reactor

- Miss Susan R. Swayne
  Documents Librarian
  Government Documents Collection
  Sacramento Public Library
  828 I Street
  Sacramento, Cal. 95814
  Rancho Seco Nuclear Generating Station

- Ms. Ann Douthett
  Reference Librarian
  San Clemente Public Library
  242 Del Mar
  San Clemente, Cal. 92672
  San Onofre Nuclear Generating Station

- Mr. Chi Su Kim
  Head—Government Documents and Maps Dept.
  Robert E. Kennedy Library
  California Polytechnic State University
  San Luis Obispo, Cal. 93407
  Diablo Canyon Nuclear Power Plant

- Mr. Gregory Cook
  Public Affairs Officer
  U.S. Nuclear Regulatory Commission, Region V
  1450 Maria Lane Suite 300
  Walnut Creek, Cal. 94596
  Vallecitos Boiling Water Reactor

COLORADO

- Miss Shirley Soenksen
  Reference Librarian
  Greeley Public Library
  City Complex Building
  919 7th Street
  Greeley, Colo. 80631
  Fort St. Vrain Nuclear Generating Station

CONNECTICUT

- Ms. Vickie Johnson
  Reference Librarian
  Russell Library
  123 Broad Street
  Middletown, Conn. 06457
  Haddam Neck Plant

- Mr. Vincent Juliano
  Director
  Waterford Public Library
  49 Rope Ferry Road
  Waterford, Conn. 06385
  Millstone Nuclear Power Station

FLORIDA

- Ms. Cheryl P. Gardieff
  Supervising Librarian
  Crystal River Public Library
  668 N. W. First Avenue
  Crystal River, Fla. 32629 Crystal River Nuclear Plant

- Miss Ruth Hansell
  Acquisitions Librarian
  Charles S. Miley Learning Resources
  Center Indian River Community
  College
  3209 Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie Plant

- Ms. Renee Pierce
  Branch Librarian
  Miami-Dade Public Library
  Homestead Branch
  700 N. Holmstead Blvd.
  Holmstead, Fla. 33030
  Turkey Point Plant

- Ms. Esther Gonzales
  Documents Librarian
  Urban and Regional Documents
  Collection
  Florida International University
  Tamiami Trail and 107th Avenue
  Miami, Fla. 33199
  Turkey Point Plant

GEORGIA

- Mrs. Wynell Bush
  Librarian
  Appling County Public Library
  301 City Hall Drive
  Baxley, Ga. 31563
  Edwin I Hatch Nuclear Plant

- Ms. Gwen Davis
  Assistant Librarian
  Burke County Library
  412 4th Street
  Waynesboro, Ga. 30830
  Alvin W. Vogtle Jr. Nuclear Plant

ILLINOIS

- Mrs. Penny A. O'Rourke
  Librarian Byron Public Library
  District 218
  W. Third Street P.O. Box 434
  Byron, Ill. 61010
  Byron Station

- Ms. Cheryle Rae Nyberg
  Assistant Law Librarian
  University of Illinois
  Law Library
  504 E. Pennsylvania Avenue
  Champaign, Ill. 61820
  Clinton Power Station

- Mrs. Betsy Taubert
  Librarian
  Vespasian Warner Public Library
  120 W. Johnson Street
  Clinton, Ill. 61727
  Clinton Power Station

- Mr. Earl R. Shumaker
  Head—Government Publications Dept.
  Founder's Memorial Library
  Northern Illinois University
  DeKalb, Ill. 60115
  Byron Station

- Mrs. Marie Hoscheid
  Head—Reference Dept.
  Moline Public Library
  504 17th Street
  Moline, Ill. 61265
  Quad Cities Station Sheffield
  Low-level Waste Burial Site

- Ms. Deborah Trotter
  Reference Assistant
  Morris Public Library
  604 Liberty Street
  Morris, Ill. 60450
  Dresden Nuclear Power Station

- Mrs. Pam Wilson Morris
  Public Library
  604 Liberty Street
  Morris, Ill. 60451
  GE-Morris Facility

- Ms. Evelyn Moyle
  Documents Librarian
  Jacobs Memorial Library
  Illinois Valley Community College
  Rural Route 1
  Oglesby, Ill. 61348
  LaSalle County Station

- Mr. Richard A. Gray
  Librarian
  Business Science and Technology
  Dept.
  Rockford Public Library
  215 North Wyman Street
  Rockford, Ill. 61101 Byron
  Station

- Mrs. Karen Stott
  Librarian
  Savanna Township Public Library
  326 Third Street
  Savanna, Ill. 61074
  Carroll County Station

- Ms. Kay Sauer
  West Chicago Public Library
  332 E. Washington Street
  West Chicago, Ill. 60185
  West Chicago Site

- Mrs. Sharon Ruda
  Director
  Government Documents Collection
  Wilmington Public Library
  201 S. Kankakee Street
  Wilmington, Ill. 60481
  Braidwood Station

- Ms. Susan G. Clark
  Adult Services Librarian
  The Memorial Library Center
  Zion-Benton Public Library District
  2400 Gabriel Avenue
  Zion, Ill. 60099
  Zion Nuclear Plant

INDIANA

- Mrs. Charlene M. Peters
  Adult Services Librarian
  Madison-Jefferson County Public
  Library
  420 W. Main Street
  Madison, Ind. 47250
  Marble Hill Nuclear Generating
  Station

IOWA

- Ms. Janice Horak
  Reference Librarian
  Cedar Rapids Public Library
  500 First Street, S.E.
  Cedar Rapids, Ia. 52401
  Duane Arnold Energy Center

KANSAS

- Ms. Vickie Bozarth
  Documents Librarian
  Government Documents Division
  William Allan White Library
  Emporia State University
  1200 Commercial Street
  Emporia, Kans. 66801
  Wolf Creek Generating Station

- Ms. Jan Brown
  Librarian
  NRC Documents Collection
  Washburn University School of Law
  Topeka, Kans. 66621
  Wolf Creek Generating Station

KENTUCKY

- Mr. Hunter Seitz
  Head—Government Documents Division
  Louisville Free Public Library
  4th and York Streets
  Louisville, Ky. 40203
  Marble Hill Nuclear Generating Station

- Ms. Beverly B. Schneider
  Library Director
  Campbell County Public Library
  4th and Monmouth Streets
  Newport, Ky. 41071
  William H. Zimmer Nuclear Power Station

LOUISIANA

- Mrs. Smittie Bolner,
  Head—Government Documents Dept.
  Troy H. Middleton Library
  Louisiana State University
  Baton Rouge, La. 70803
  River Bend Station

- Mr. Kenneth E. Owen
  Head—Louisiana Collection
  Earl K. Long Library
  University of New Orleans
  Lakefront Drive
  New Orleans, La. 70148
  Waterford Generating Station

MAINE

- Mrs. Barbara Shelton
  Librarian
  Wiscasset Public Library
  High Street P.O. Box 367
  Wiscasset, Me. 04578
  Maine Yankee Atomic Power Plant

MARYLAND

- Ms. Marcia G. Hammett
  Reference Librarian
  Calvert County Public Library
  Fourth Street P.O. Box 300
  Prince Frederick, Md. 20678
  Calvert Cliffs Nuclear Power Plant

MASSACHUSETTS

- Mrs. Margaret E. Howland
  Director of Learning Resources
  Library/Learning Resource Center
  Greenfield Community College
  1 College Drive
  Greenfield, Mass. 01301
  Yankee Rowe Nuclear Power Station

- Ms. Grace E. Karbott
  Library Associate
  Plymouth Public Library
  11 North Street
  Plymouth, Mass. 02360
  Pilgrim Nuclear Power Station

MICHIGAN

- Mr. Gig Stewart
  Library Director
  North Central Michigan College
  1515 Howard Street
  Petoskey, Mich. 49770
  Big Rock Point Nuclear Plant

- Ms. Carol Juth
  Reference Librarian
  Van Zoeren Library
  Hope College
  Holland, Mich. 49423
  Palisades Nuclear Plant

- Ms. Sandra Krchmar
  Reference Librarian
  Grace A. Dow Memorial Library
  1710 W. St. Andrews Road
  Midland, Mich. 48640
  Midland Plant

- Ms. Janice Murphy
  Reference/Government Documents Coordinator
  Ellis Reference and Information Center
  Monroe County Library System
  3700 S. Custer Road
  Monroe, Mich. 48161
  Enrico Fermi Atomic Power Plant

- Ms. Bea Rodgers
  Library Assistant
  Maude Preston Palenske Memorial Library
  500 Market Street St.
  Joseph, Mich. 49085
  Donald C. Cook Nuclear Power Plant

MINNESOTA

- Mr. Thomas Smisek
  Librarian
  Technology and Science Dept.
  Minneapolis Public Library
  300 Nicollet Mall
  Minneapolis, Minn. 55401
  Monticello Nuclear Generating Plant
  Prairie Island Nuclear Generating Plant

MISSISSIPPI

- Mrs. Gayle Keefe
  Library Technical Assistant
  George M. McLendon Library
  Hinds Junior College
  Main Street
  Raymond, Miss. 39154
  Grand Gulf Nuclear Plant

MISSOURI

- Mrs. Evelyn Hillard
  Public Services Librarian
  Callaway County Public Library
  709 Market Street
  Fulton, Mo. 65251
  Callaway Plant

- Mr. B. J. Johnston
  Government Publications Librarian
  John M. Olin Library
  Washington University
  Skinker and Lindell Boulevards
  St. Louis, Mo. 63130
  Callaway Plant

NEBRASKA

- Mrs. Loy Mowery
  Director
  Auburn Public Library
  1118 15th Street P.O. Box 324
  Auburn, Neb. 68305
  Cooper Nuclear Station

- Mr. William Kendra
  Business, Science and Technology Dept.
  W. Dale Clark Library
  215 S. 15th Street
  Omaha, Neb. 68102
  Fort Calhoun Station

NEW HAMPSHIRE

● Ms. Pamela Gjettum
Director
Exeter Public Library
Front Street
Exeter, N.H. 03833
Seabrook Nuclear Station

NEW JERSEY

● Miss Joanne L. Owens
Librarian
Pennsville Public Library
190 S. Broadway
Pennsville, N.J. 08070
Hope Creek Nuclear Generating Station

● Ms. Elizabeth C. Fogg
Director
Salem Free Public Library
112 West Broadway
Salem, N.J. 08079
Salem Nuclear Generating Station

● Ms. Lois J. Brown
Reference Librarian
Reference Dept.
Ocean County Library
101 Washington Street
Toms River, N.J. 08753
Oyster Creek Nuclear Power Plant

NEW YORK

● Mr. Stanley Zubowski
Buffalo and Erie County Public Library
Lafayette Square
Buffalo, N.Y. 14203
West Valley Demonstration Project

● Mr. Sol Becker
Chief Librarian
Public Health Library
New York City Dept. of Health
125 Worth Street
New York, N.Y. 10013
Columbia University Research Reactor

● Mrs. Adele Mangano
Library Technician
Reference and Documents Dept.
Penfield Library
State University of New York
Oswego, N.Y. 13126
Nine Mile Point Nuclear Station
James A. Fitzpatrick Nuclear Power Plant

● Ms. Cynthia A. Dana
Senior Library Clerk
Business and Social Science Division
Rochester Public Library
115 South Avenue
Rochester, N.Y. 14610
Robert E. Ginna Nuclear Power Plant

● Ms. Cathy McGowan
Librarian
Shoreham-Wading River Public Library
Route 25A
Shoreham, N.Y. 11786
Shoreham Nuclear Power Station

● Ms. Annette Gernatt
Town of Concord
Public Library
23 N. Buffalo Street
Springfield, N.Y. 14141
West Valley Demonstration Project

● Mr. Oliver F. Swift
Municipal Reference Librarian
White Plains Public Library
100 Martine Avenue
White Plains, N.Y. 10601
Indian Point Station

NORTH CAROLINA

● Ms. Dawn Hubbs
J. Murrey Atkins Library
University of North Carolina at Charlotte
Charlotte, N.C. 28223
William B. McGuire Nuclear Station

● Mrs. Joe Ann Stephens
Reference Librarian
Richard B. Harrison Library
1313 Bern Avenue
Raleigh, N.C. 27610
Shearon Harris Nuclear Power Plant

● Ms. Emma Myles
Library Assistant
Brunswick County Library
109 W. Moore Street
Southport, N.C. 28461
Brunswick Steam Electric Plant

OHIO

● Ms. Vera E. Ehas
Public Services Librarian
Clermont County Public Library
180 South Third Street
Batavia, Ohio 45103
William H. Zimmer Nuclear Power Station

● Mrs. Shirley Morgan
Reference Librarian
Perry Public Library
3753 Main Street
Perry, Ohio 44081
Perry Nuclear Power Plant

● Mrs. Julia Baldwin
Documents Librarian
Government Documents Collection
William Carlson Library
University of Toledo
2801 West Bancroft Avenue
Toledo, Ohio 43606
Davis-Besse Nuclear Power Station

OKLAHOMA

● Mrs. O.J. Grosclaude
Sallisaw City Library
111 North Elm
Sallisaw, Okla. 74995
Kerr-McGee Corporation

OREGON

● Mr. Jim Takita
Head—Social Science and Science
Dept. Government Documents
Collection Library Association of Portland
801 S. W. 10th Avenue
Portland, Ore. 97205 Trojan Nuclear Plant

PENNSYLVANIA

● Ms. Nancy Luczinger
Reference Librarian
B.F. Jones Memorial Library
663 Franklin Avenue
Aliquippa, Pa. 15001
Beaver Valley Power Station
Shippingport Project

● Ms. Datz Apollo
Memorial Library
219 N. Pennsylvania
Apollo, Pa. 15613
Babcock & Wilcox Parks Township
Fuel Facility

● Mr. Phil Hearne
Librarian
Dauphin Library System
101 Walnut Street
Harrisburg, Pa. 17101
Three Mile Island Nuclear Station

● Mr. Lawrence H. Peterson
Reference Librarian
Government Publications Section
State Library of Pennsylvania
Walnut Street and Commonwealth
Avenue
Box 1601
Harrisburg, Pa. 17105
Three Mile Island Nuclear Station
Peach Bottom Atomic Power
Station Fulton Nuclear Station

● Mr. William A. Felker
Reference and Information Manager
Government Publications Dept.
Free Library of Philadelphia
19th and Vine Streets
Philadelphia, Pa. 19103
Three Mile Island Nuclear Station
Limerick Generating Station

● Mrs. Julia Albright
Interlibrary Loan Librarian
Pottstown Public Library
500 High Street
Pottstown, Pa. 19464
Limerick Generating Station

● Mr. Ernest Fuller
NRC Materials Aide
Saxton Community Library
911 Church Street
Saxton, Pa. 16678
Saxton Experimental Reactor Project

● Ms. Diane H. Smith
Head—Government Documents
Pattee Library
Room C 207
Pennsylvania State University
University Park, Pa. 16802
Three Mile Island Nuclear Station
Susquehanna Steam Electric
Station Beaver Valley Power
Station

● Ms. Lisa Stegmueller
Reference Librarian
Reference Dept.
Osterhout Free Library
71 S. Franklin Street
Wilkes-Barre, Pa. 18701
Susquehanna Steam Electric Station

● Mr. David Van de Streek
Librarian
Pennsylvania State University
York Campus
1031 Edgecomb Avenue
York, Pa. 17403
Three Mile Island Nuclear Station

RHODE ISLAND

● Ms. Ann Crawford
Cross Mill Public Library
Old Post Road
Charlestown, R.I. 02813
Wood River Junction

● Mr. Tom Reynolds
Univ. of Rhode Island
University Library, Govt. Publications
Section
Kingston, R.I. 02881
Wood River Junction

SOUTH CAROLINA

● Miss Ava G. Black
Head Librarian
Barnwell County Public Library
Hagood Avenue
Barnwell, S.C. 29812
Barnwell Reprocessing Plant
Barnwell Low-level Waste Burial
Site

● Ms. Maureen Harris
Head—Public Documents Collection
Robert M. Cooper Library
Clemson University
Clemson, S.C. 29631
Oconee Nuclear Plant

● Ms. Mary Toll
Reference Librarian
Technical Services Dept.
South Carolina State Library
1500 Senate Street
Columbia, S.C. 29201
Catawba Nuclear Station

● Mrs. Jane Mason
Librarian
Nuclear Information Depository
Hartsville Memorial Library
220 N. Fifth Street
Hartsville, S.C. 29550
H. B. Robinson Plant

● Mrs. Mary Mallaney
Assistant Reference Librarian
York County Library
138 East Black Street
P.O. Box 10032
Rock Hill, S.C. 29730
Catawba Nuclear Station

● Ms. Joyce McCall
Librarian
Oconee County Library
501 W. South Broad Street
Walhalla, S.C. 29691
Oconee Nuclear Plant

● Ms. Sarah D. McMaster
Director
Fairfield County Library
Garden and Washington Streets
Winnsboro, S.C. 29180
Virgil C. Summer Nuclear Station

TENNESSEE

● Ms. Patricia Maroney
Head—Business, Science and
Technology Dept. Chattanooga-
Hamilton County Library
1001 Broad Street
Chattanooga, Tenn. 37402
Sequoyah Nuclear Plant Watts Bar
Nuclear Plant

● Ms. Carol A. Goris
Reference Assistant
Lawson McGhee Public Library
500 W. Church Avenue
Knoxville, Tenn. 37902
Clinch River Breeder Reactor
Project

● Mrs. Carol P. Cooper
Library Assistant
Depository Collection Reference Dept.
Oak Ridge Public Library
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Appendix 4

Regulations and Amendments—Fiscal Year 1985

The regulations of the Nuclear Regulatory Commission are contained in Title 10, Chapter 1, of the Code of Federal Regulations. Effective and proposed regulations concerning licensed activities, and certain policy statements related to them, which were published in the Federal Register during fiscal year 1985, are described briefly below.

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Training and Qualifications of Civilian Nuclear Power Plant Personnel and Operator's Licenses—Part 55

On October 24, 1984 (49 FR 42693), the NRC published an amendment to its regulations, effective November 23, 1984, governing the training and qualifications of civilian nuclear power plant personnel. The amendment conforms the literal language of the regulations to the long-standing agency practice of treating the satisfactory completion of an NRC approved program for training reactor operators as the equivalent of actual operating experience at a reactor.

Environmental Qualification of Electrical Equipment; Removal of June 30, 1982 Deadline—Part 50

On November 19, 1984 (49 FR 45571), the NRC published an amendment to its regulations, effective immediately, that deletes a June 30, 1982, deadline for the environmental qualification of electrical equipment from power plant operating licenses. The final rule specifies the schedule for environmental qualification that licensees of operating power plants are expected to meet.

Export and Import of Nuclear Equipment and Material—Part 110

On December 3, 1984 (49 FR 47191), the NRC published an amendment to its regulations, effective January 2, 1985, pertaining to the export and import of nuclear equipment and material. The amended regulations expand the authority to export nonsensitive nuclear equipment and minor quantities of nuclear material without applying for and obtaining a specific NRC license authorizing the action.

Hydrogen Control Requirements—Part 50

On January 25, 1985 (50 FR 3498), the NRC published an amendment to its regulations that improves hydrogen control capability for boiling water reactors with MARK III containments and for pressurized water reactors with ice condenser containments. The amendments, effective February 25, 1985, require improved hydrogen control systems that can handle large amounts of hydrogen during and after an accident.

Criteria and Procedures for Determining the Adequacy of Available Spent Nuclear Fuel Storage Capacity—Parts 1 and 53

On February 11, 1985 (50 FR 5548), the NRC published an amendment to its regulations, effective March 13, 1985, that establishes procedures and criteria for determining whether a person owning or operating a civilian nuclear power reactor cannot reasonably provide adequate spent nuclear fuel storage capacity. This determination is necessary before the Department of Energy may enter into contractual arrangements to provide interim Federal storage capacity for limited amounts of spent nuclear fuel.

Amended Material Control and Accounting Requirements for Special Nuclear Material of Low Strategic Significance—Parts 70 and 74

On February 25, 1985 (50 FR 7575), the NRC published an amendment to its regulations concerning material control and accounting (MC&A) requirements for licensees possessing and using quantities larger than one effective kilogram of special nuclear material of low strategic significance. The amendments, effective March 27, 1985, reform the MC&A regulations for fuel cycle facilities by establishing a grading of requirements between those applicable to more strategically significant forms of special nuclear material and those applicable to low enriched uranium.

Implementation of the Convention on Physical Protection of Nuclear Material Parts 40, 70, 73 and 110

On March 28, 1985 (50 FR 12221), the NRC published an amendment to its regulations to bring them into accord with the provisions of the Convention on the Physical Protection of Nuclear Material. The amendments, which become effective 30 days after the 21st country ratifies the Convention, will result in strengthened protection of shipments of Convention-defined materials during international transport.

Exceptions to Notice and Comment Rulemaking Procedures—Part 2

On April 2, 1985 (50 FR 13006), the NRC published an amendment to its rules of practice, effective May 3, 1985, that revised the Commission's rulemaking procedures. The final rule clarifies the Commission's use of the exceptions to notice and comment rulemaking contained in the Administrative Procedure Act.

Regional Nuclear Materials Licensing For Certain Federal Facilities Parts 30, 40, and 70

On April 15, 1985 (50 FR 14692), the NRC published an amendment to its regulations, effective April 1, 1985, that further decentralized the licensing process for materials licensees. The amendment extends to the Regional Offices the same licensing authority for certain Federal licensees that they possess for non-Federal activities.

Emergency Planning and Preparedness—Part 50

On May 8, 1985 (50 FR 19323), the NRC published an amendment, effective immediately, that revised its emergency planning and preparedness regulations for nuclear power reactors. This amendment, made in response to a decision by the United States Court of Appeals for the District of Columbia Circuit, removes the provision stating that emergency preparedness exercises are not required for any initial licensing decision.

Delegation of Subpoena Authority—Part 1

On May 20, 1985 (50 FR 20741), the NRC published an amendment to its regulations that reflected the Commission's decision to delegate the authority to issue subpoenas to the Office of Investigations where necessary or appropriate for the conduct of investigations. This amendment, effective June 19, 1985, permits the Office of Investigations to issue a subpoena independently during the course of an investigation.

Export of Reprocessing Plant Components—Part 110

On May 20, 1985 (50 FR 20742), the NRC published an amendment to its regulations that further clarified the list of components considered especially designed or prepared for use in a nuclear fuel reprocessing plant and therefore subject to the Commission's export licensing authority. This amendment, effective May 21, 1985, implements a decision of the multilateral Non-Proliferation Treaty Exporters Committee.

Government in the Sunshine Act Regulations—Part 9

On May 21, 1985 (50 FR 20889), the NRC published an interim rule, effective immediately, to conform its definition of "meeting" under the Government in the Sunshine Act to the statutory intent as clarified in a recent Supreme Court decision. As a result of this change, background briefings and generalized discussions of agency business are not considered "meetings" for Sunshine Act purposes.

Update of NRC Addresses and Copying Charges for Environmental Documents Parts 1 and 51

On May 22, 1985 (50 FR 21036), the NRC published an amendment to its regulations, effective immediately, that updated the addresses of NRC's principal offices and conformed the charges for the reproduction of environmental documents at the NRC's Public Document Room to the charges found in 10 CFR Part 9. Charges for the Production of Records—Part 9

On June 18, 1985 (50 FR 25204), the NRC published an amendment to its regulations, effective immediately, that revised the charges for copying records available in the NRC's Public Document Room.

Conduct of Employees; Minor Amendments—Part 0

On June 21, 1985 (50 FR 25697), the NRC published an amendment to its standards of conduct for employees. The amendment, effective immediately, codifies provisions of the Ethics in Government Act of 1978 relating to the reporting of assets by senior NRC officials. This amendment also makes several additional minor changes to the regulations governing the communication of scientific or technological information to the NRC by former employees, the acceptance of gifts, meals, and entertainment from foreign governments, and the publication of the prohibited securities list.

Public Records; Freedom of Information Act; Granting Appeals—Part 9

On July 2, 1985 (50 FR 27214), the NRC published an amendment to its Freedom of Information Act (FOIA) regulations. The amendment, effective July 1, 1985, provides that appeals from an initial denial of a FOIA request be decided by the Secretary of the Commission, with the advice and concurrence of the Office of the General Counsel, instead of being decided by the Commission.

Disposal of High-Level Radioactive Wastes in Geologic Repositories—Part 60

On July 22, 1985 (50 FR 29641), the NRC published an amendment to its regulations, effective immediately, for the disposal of high-level radioactive wastes in geologic repositories. This amendment sets out specific criteria for the disposal of high-level radioactive wastes within the unsaturated zone. This action is necessary to assure that NRC regulations address considerations relevant to all geologic repositories.

Analysis of Potential Pressurized Thermal Shock Events—Part 50

On July 23, 1985 (50 FR 29937), the NRC published an amendment to its regulations, effective immediately, for light water nuclear power plants. The amendment establishes a screening criterion related to the fracture resistance of pressurized water reactor vessels during pressurized thermal shock events, requires analyses and schedules for implementation of flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion, and requires that detailed safety operations be performed before plant operation beyond the screening criterion will be considered.

Minor Clarifying Amendments; Definitions—Parts 1, 20, and 50

On August 23, 1985 (50 FR 34085), the NRC published an amendment to its regulations pertaining to the domestic licensing of production and utilization facilities. The amendment, effective immediately, rearranged the definitions section from a listing by alphabetical designators to an undesignated alphabetical listing by term.

Physician's Use of Radioactive Drugs—Part 35

On September 10, 1985 (50 FR 36866), the NRC published an amendment to its regulations that allows medical licensees to use certain radioactive drugs for specified clinical purposes. The amendment, effective immediately, allows the use of specified drugs for particular procedures without requiring a physician or hospital to apply for an amendment to their license.

Access to and Protection of National Security Information and Restricted Data Parts 25 and 95

On September 11, 1985 (50 FR 36983), the NRC published an amendment to its regulations incorporating an exception to personal security background investigation requirements for access to certain Communications Security information. The amendments, effective October 11, 1985, also provide a procedure to ensure that a licensee obtains prior NRC approval before making any substantive changes to the licensee's security plan.

Procedures for Production or Disclosure of Records or Information in Response to Subpoenas or Demands of Courts or Other Authorities—Part 9

On September 17, 1985 (50 FR 37642), the NRC published an amendment to its regulations that prescribes procedures for the production of documents or the disclosure of information in response to subpoenas or demands of courts or other judicial or quasi-judicial authorities in State and Federal proceedings. The amendment, effective October 17, 1985, clarifies Commission procedures regarding subpoenas or other judicial or quasi-judicial demands on NRC employees to produce NRC records or disclosure information and ensures that the responsi-bility for determining the response to the demands is placed on the appropriate Commission official.

Revision of Backfitting Process for Power Reactors—Parts 2 and 50

On September 20, 1985 (50 FR 38097), the NRC published an amendment to its regulations concerning "backfitting," a process which can include both plant specific and generic changes applied to one or more classes of power reactors. The amendment, effective October 21, 1985, establishes standards and an agency discipline for the future management of backfitting for power reactors.

Codes and Standards for Nuclear Power Plants—Part 50

On September 26, 1985 (50 FR 38970), the NRC published an amendment to its regulations incorporating by reference the Winter 1982 Addenda, Summer 1983 Addenda, Winter 1983 Addenda, Summer 1984 Addenda, and 1983 Edition of Section III, Division I of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and the Winter 1982 Addenda, Summer 1983 Addenda, and 1983 Edition of Section XI, Division I of the ASME Code. The amendment, effective October 28, 1985, permits the use of improved methods for construction and inservice inspection of nuclear power plants.

Criteria and Procedures for Determining Eligibility for Access to or Control Over Special Nuclear Material—Part 11

On September 27, 1985 (50 FR 39076), the NRC published an amendment to its regulations allowing the use of information on file with the Federal government for individuals possessing current active clearances based on equivalent investigations. The amendment, effective October 28, 1985, applies to initial special nuclear material "U" and "R" access authorization applications and for the renewal of "U" access authorizations. The amendment eliminates unnecessary administrative and investigative costs to licensees for affected individuals and reduces NRC administrative costs.

REGULATIONS AND AMENDMENTS PROPOSED

Industrial Radiography Radiation Surveys and Licensee's Performance Inspection Program—Part 34

On October 4, 1984 (49 FR 39168), the NRC published a notice of proposed rulemaking that would amend its regulations pertaining to industrial radiography. The proposed rule would require that an industrial radiography licensee perform an additional survey of any radiography device any time the device is put into storage and maintain a record of this storage survey in place of the previously required record of the survey of the device made after the last exposure. The proposed rule would also require that each license application describe the program the licensee will use to evaluate the performance of each radiographer and radiographer's assistant at intervals not exceeding 3 months to ensure that they are following the Commission's regulatory requirements and the licensee's operating and emergency procedures.

Changes in Property Insurance Requirements for NRC Licensed Nuclear Power Plants—Part 50

On November 8, 1984 (49 FR 44645), the NRC published a notice of proposed rulemaking that would amend regulations requiring that licensees maintain substantial amounts of on-site property insurance to assist in the decontamination of their licensed reactors. The proposed rule would increase the amount of insurance required and impose a decontamination priority on any proceeds from the insurance.

Uranium Mill Tailings Regulations: Conforming NRC Requirements to EPA Standards—Part 40

On November 26, 1984 (49 FR 46418), the NRC published a notice of proposed rulemaking that would amend its regulations governing the disposal of uranium mill tailings. The proposed changes are intended to conform existing NRC regulations to the regulations published by the Environmental Protection Agency for the protection of the environment from these wastes.

Operator's Licenses and Conforming Amendments—Parts 50 and 55

On November 26, 1984 (49 FR 46428), the NRC published a notice of proposed rulemaking concerning its operator licensing regulations. The proposed rule would (1) clarify the regulations for the issuance of licenses to operators and senior operators; (2) revise the requirements and scope of written examinations and operating tests for operators and senior operators including a requirement for a simulation facility; (3) codify procedures for the administration of requalification examinations; and (4) describe the form and content for operator license applications.

Revision of Backfitting Process for Power Reactors—Parts 2 and 50

On November 30, 1984 (49 FR 47034), the NRC published a notice of proposed rulemaking that addresses the "backfitting" issue. The proposed rule would establish requirements for the long-term management of its review process for the implementation of new regulatory requirements on power reactors.

Emergency Planning and Preparedness for Production and Utilization Facilities Part 50

On December 21, 1984 (49 FR 49640), the NRC published a notice of proposed rulemaking that would amend its regulations concerning emergency planning and preparedness for nuclear power reactors. The proposed regulations would make previous Commission rulings that consideration of potential impacts of earthquakes an emergency planning for nuclear reactor sites is not required explicit in its regulations.

Criteria for Reopening Records in Formal Licensing Proceedings—Part 2

On December 27, 1984 (49 FR 50189), the NRC published a notice of proposed rulemaking that would codify NRC case law criteria for reopening a closed evidentiary record in a formal licensing proceeding and would further specify the documentary bases for motions to reopen. The proposed rule is intended to facilitate proper and timely consideration of motions by adjudicatory boards while maintaining fairness to all other parties to a proceeding.

Disposal of High-Level Radioactive Waste in Geologic Repositories; Amendments to Licensing Procedures—Part 60

On January 17, 1985, (50 FR 2579) the NRC published a notice of proposed rulemaking that would revise procedures with respect to NRC reviews of license applications for disposal of high-level radioactive waste in geologic repositories. The proposed revisions reflect provisions of the Nuclear Waste Policy Act of 1982 relating to site characterization and the participation of States and Indian tribes in the process of siting, licensing, and developing disposal facilities.

Decommissioning Criteria for Nuclear Facilities—Parts 30, 40, 50, 51, 70 and 72

On February 11, 1985 (50 FR 5600), the NRC published a notice of proposed rulemaking that would set forth technical and financial criteria for decommissioning licensed facilities. The proposed rule is intended to ensure that all licensed facilities will be decommissioned in a safe and timely manner and that adequate licensee funds will be available for decommissioning.

Access to and Protection of National Security Information and Restricted Data Parts 25 and 95

On March 13, 1985 (50 FR 10064), the NRC published a notice of proposed rulemaking that would amend its regulations to incorporate a recently approved exception to the personnel security background investigation for access to certain Communications Security information. The proposed rule would also provide a procedure to ensure that a licensee obtains prior NRC approval before making any substantive changes to the licensee's security plan.

Communications Procedures Amendments—Part 50

On March 26, 1985 (50 FR 11884), the NRC published a notice of proposed rulemaking that would establish procedures for submitting correspondence, reports, applications, or other written communications concerning the domestic licensing of production and utilization facilities. The proposed changes are intended to resolve problems that have developed in the submission of applications and reports by indicating the correct mailing address for written communications and specifying the number of copies required to facilitate NRC action.

Licenses and Radiation Safety Requirements For Well-Logging Operations Parts 19, 20, 21, 30, 39, 40, 51, 70, 71, and 150

On April 8, 1985 (50 FR 13797), the NRC published a notice of proposed rulemaking that would specify radiation safety requirements for the use of licensed material in well-logging operations. The proposed rule is intended to consolidate regulations applicable to well-logging operations, provide uniform safety requirements in NRC and Agreement State regulations, and reduce the risks of accidents involving the rupture of a radioactive source in well-logging operations.

Criteria for an Extraordinary Nuclear Occurrence—Part 140

On April 9, 1985 (50 FR 13978), the NRC published a notice of proposed rulemaking that would revise the criteria for an "extraordinary nuclear occurrence". The proposed rule is intended to simplify the administrative criteria used in making an extraordinary nuclear occurrence determination and avoid the problems encountered by applying the existing criteria to the accident at Three Mile Island.

Physician's Use of Radioactive Drugs—Part 35

On April 22, 1985 (50 FR 15752), the NRC published a notice of proposed rulemaking that would amend its regulations to allow medical licensees to use certain radioactive materials for specified clinical procedures. The proposed rule would allow the use of specified drugs for particular procedures without requiring a physician or hospital to apply for an amendment to their license.

Specific Exemptions—Part 50

On April 26, 1985 (50 FR 16056) the NRC published a notice of proposed rulemaking that would clarify the standards that will be applied by the Commission when it considers whether to grant exemptions from the regulatory requirements codified in 10 CFR Part 50.

Material Balance Reports—Parts 40, 70, and 150

On May 10, 1985 (50 FR 19695), the NRC published a notice of proposed rulemaking concerning the submission of source material and special nuclear material inventory reports. The proposed rule would reduce the reporting burden for specific licensees without adversely affecting the domestic safeguards program or the ability to satisfy existing international commitments.

Codes and Standards for Nuclear Power Plants—Part 50

On May 17, 1985 (50 FR 20574), the NRC published a notice of proposed rulemaking that would incorporate by reference the Winter 1982 Addenda, Summer 1983 Addenda, Winter 1983 Addenda, Summer 1984 Addenda, and 1983 Edition of Section III, Division I, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and the Winter 1982 Addenda, Summer 1983 Addenda, and 1983 Edition of Section XI, Division I of the ASME Code. The proposed rule would permit the use of improved methods for construction and inservice inspection of nuclear power plants.

Adjudications; Special Procedures for Resolving Conflicts Concerning the Disclosure or Nondisclosure of Information—Part 2

On May 22, 1985 (50 FR 21072), the NRC published a notice of proposed rulemaking that would amend its rules of practice. The proposed rule would provide special procedures for resolving conflicts concerning the disclosure or nondisclosure of information relating to an NRC investigation or inspection not yet concluded or which would reveal the identity of a confidential informant considered relevant and material to an adjudication.

Modification of General Design Citerion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures—Part 50

On July 1, 1985 (50 FR 27006), the NRC published a notice of proposed rulemaking that would modify General Design Criterion 4 to allow demonstration of piping integrity by analyses to serve as a basis for excluding dynamic effects associated with certain pipe ruptures. These analyses constitute what is commonly referred to as the "leak before break" concept.

Medical Use of Byproduct Material—Parts 30, 31, 32, 35, and 40

On July 26, 1985 (50 FR 30616), the NRC published a notice of proposed rulemaking that would modify the process for licensing and regulating the medical use of radioactive byproduct material. The proposed rule would consolidate and clarify the essential radiation safety requirements applicable to the medical uses of byproduct materials. The proposed rule would also provide licensees flexibility in the updating of their day-to-day radiation safety procedures.

Minor Clarifying Amendments—Part 9

On August 1, 1985 (50 FR 31192), the NRC published a notice of proposed rulemaking that would clarify its regulations pertaining to the availability of records under the Freedom of Information Act by conforming them to reflect existing case law and long-standing agency practice. The proposed rule would also conform reproduction costs charged at the Public Document Room and other NRC offices for publicly available documents.

Changes to Safeguards Reporting Requirements—Parts 70, 72, 73, and 74

On August 27, 1985 (50 FR 34708), the NRC published a notice of proposed rulemaking that would clarify reporting requirements for NRC licensees and would improve the NRC safeguards event data base by requiring more uniform safeguards event reports. The proposed rule would eliminate unnecessary reporting and result in a more uniform and detailed reporting and data analysis system which will provide feedback to the industry for improving safeguards systems.

ADVANCE NOTICES OF PROPOSED RULEMAKING

Uranium Mill Tailings Regulations: Ground Water Protection and Other Issues Part 40

On November 26, 1984 (49 FR 46425), the NRC published an advance notice of proposed rulemaking announcing its intent to consider further amendments to its uranium mill tailings regulations. This future rulemaking proceeding is intended to incorporate ground water protection provisions and other requirements established by the Environmental Protection Agency for similar hazardous wastes into NRC regulations.

Financial Responsibility Requirements Applicable to NRC Licensees for Cleanup of Accidental and Unexpected Releases of Radioactive Materials— Parts 30, 40, 61, 70, and 72

On June 7, 1985 (50 FR 23960), the NRC published an advance notice of proposed rulemaking announcing its intent to consider requiring certain materials licensees to demonstrate that they possess adequate financial means to pay for cleanup of accidental releases of radioactive materials. This document invites advice and recommendations on the scope of this potential rulemaking, as well as the availability and cost to licensees of the various forms of financial assurance.

Certification of Industrial Radiographers—Part 34

On September 19, 1985 (50 FR 38011), the NRC published a document withdrawing an advance notice of proposed rulemaking requesting comments on a suggested requirement that all industrial radiographers be certified by a third party approved by the NRC.

Appendix 5

Regulatory Guides—Fiscal Year 1985

NRC regulatory guides describe methods for implementing specific parts of the Commission's regulations and, in some cases, describe techniques used by the staff in evaluating specific problems or postulated accidents. Guides also may advise applicants regarding information the NRC staff needs in reviewing applications for permits and licenses.

Comments on the guides are encouraged, and the guides are revised whenever appropriate to reflect new information or experience. NRC issues the guides for public comment in draft form before they have received complete staff review and an official staff position has been established.

Once issued, regulatory guides may be withdrawn when superseded by Commission regulations, when equivalent recommendations have been incorporated in applicable approved codes and standards, or when changes make them obsolete.

When guides are issued, revised, or withdrawn, notices are placed in the *Federal Register.*

To reduce the burden on the taxpayer, the NRC has made arrangements for the sale of active regulatory guides by both the U.S. Government Printing Office (on an individual guide basis) and the National Technical Information Service (on a standing order basis). Draft guides issued for public comment receive free distribution. NRC licensees receive, at no cost, pertinent draft and active regulatory guides as they are issued.

The following guides were issued, revised, or withdrawn during the period October 1, 1984, to September 30, 1985.

Division 1—Power Reactor Guides

1.28	Quality Assurance Program Requirements (Design and Construction) (Revision 3)
1.46	WITHDRAWN. Protection Against Pipe Whip Inside Containment
1.48	WITHDRAWN. Design Limits and Loading Combinations for Seismic Category I Fluid System Components
1.84	Design and Fabrication Code Case Acceptability—ASME Section III, Division 1 (Revision 23)
1.85	Materials Code Case Acceptability—ASME Section III, Division 1, (Revision 23)
1.147	Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1 (Revision 4)

Division 2—Research and Test Reactor Guides

NONE

Division 3—Fuels and Materials Facilities Guides

3.55	Standard Format and Content for the Health and Safety Sections of License Renewal Applications for Uranium Hexafluoride Production

Division 4—Environmental and Siting Guides

NONE

Division 5—Materials and Pland Protection Guides

5.6	WITHDRAWN. Standard Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Plutonium Dioxide Powders and Pellets and Nuclear-Grade Mixed Oxides ([U, Pu]O+J 2)
5.16	WITHDRAWN. Standard Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Nuclear-Grade Plutonium Nitrate Solutions and Plutonium Metal (Revision 1)
5.19	WITHDRAWN. Methods for the Accountability of Plutonium Nitrate Solutions
5.40	WITHDRAWN. Methods for the Accountability of Plutonium Dioxide Powder
5.47	WITHDRAWN. Control and Accountability of Plutonium in Waste Material

Division 6—Product Guides

NONE

Division 7—Transportation Guides

NONE

Division 8—Occupational Health Guides

NONE

Division 9—Antitrust and Financial Review Guides

NONE

Division 10—General Guides

NONE

Draft Guides

Division 1

OL 401-5	Proposed Revision 2 to Regulatory Guide 1.134, Medical Evaluation of Nuclear Facility Personnel Requiring Operator Licenses
OL 402-5	Proposed Revision 1 to Regulatory Guide 1.149, Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations
OL 403-5	Third Proposed Revision 2 to Regulatory Guide 1.8, Qualification and Training of Personnel for Nuclear Power Plants

Division 3

CE 308-4	Proposed Revision 1 to Regulatory Guide 3.52, Standard Format and Content for the Health and Safety Sections of License Renewal Applications for Uranium Processing and Fuel Fabrication
CE 309-4	General Guidance for Designing, Testing, Operating, and Maintaining Emission Control Devices at Uranium Mills
CE 404-4	Proposed Revision 2 to Regulatory Guide 3.4, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
CE 407-4	Criticality Safety for Handling, Storing, and Transporting LWR Fuel Outside Reactors
ES 401-4	Onsite Meteorological Measurement Program for Uranium Recovery Facilities—Data Acquisition and Reporting

Division 4

CE 401-4	Proposed Revision 1 to Regulatory Guide 4.16, Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants
WM 404-4	Proposed Revision 1 to Regulatory Guide 4.17, Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories

Division 5

KG 302-4	Vital Islands, Protection of Physical Security Equipment, and Key and Lock Controls

Division 10

FC 401-4	Proposed Revision 2 to Regulatory Guide 10.6, Guide for the Preparation of Applications for the Use of Sealed Sources and Devices for Performing Industrial Radiography
FC 402-4	Second Proposed Revision 1 to Regulatory Guide 10.9, Guide for the Preparation of Applications for Licenses for the Use of Self-Contained Dry Source-Storage Irradiators
FC 403-4	Guide for the Preparation of Applications for Licenses for the Use of Panoramic Dry Source-Storage Irradiators, Self-Contained Wet Source-Storage Irradiators, and Panoramic Wet Source-Storage Irradiators
FC 404-4	Guide for the Preparation of Applications for Licenses for the Use of Sealed Sources in Nonportable Gauging Devices
FC 405-4	Guide for the Preparation of Applications for Licenses for the Use of Sealed Sources in Gas Chromatography Devices and X-Ray Fluorescence Analyzers
FC 406-4	Guide for the Preparation of Applications for Licenses and Approvals To Authorize Distribution of Various Items to Group Medical Licensees
FC 407-4	Guide for the Preparation of Applications for Licenses for the Use of Sealed Sources in Portable Gauging Devices
FC 408-4	Proposed Revision 2 to Regulatory Guide 10.5, Guide for the Preparation of Applications for Type A Licenses of Broad Scope
FC 409-4	Proposed Revision 2 to Regulatory Guide 10.4, Guide for the Preparation of Applications for Licenses To Process Source Material
FC 410-4	Guide for the Preparation of Applications for Nuclear Pharmacy Licenses
FC 411-4	Guide for the Preparation of Applications for the Use of Radioactive Materials in Servicing Preregistered Gauges, Measuring Devices, and Sealed Sources Used in Such Devices
FC 412-4	Guide for the Preparation of Applications for Licenses for the Use of Radioactive Materials in Leak-Testing Services
FC 413-4	Guide for the Preparation of Applications for Licenses for the Use of Radioactive Materials in Calibrating Radiation Survey and Monitoring Instruments
FC 415-4	Proposed Revision 2 to Regulatory Guide 10.8, Guide for the Preparation of Applications for Medical Programs

Appendix 6

Nuclear Electric Generating Units in Operation Or Under Construction

(As of December 31, 1985)

The following listing includes nuclear power reactor electrical generating units which were in operation or under construction or under construction in the United States as of December 31, 1985, representing a total capacity of approximately 116,000 MWe. Reactor types are indicated as follows: BWR—boiling water reactor, PWR—pressurized water reactor, HTGR—high temperature gas-cooled reactor. Plant status is indicated as follows: OL—has operating license, CP—has construction permit. The dates for operation are either actual or as scheduled by the utilities as of December 31, 1985.

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
ALABAMA						
Decatur	Browns Ferry Nuclear Power Plant Unit 1	1,065	BWR	OL 1983	Tennessee Valley Authority	1974
Decatur	Browns Ferry Nuclear Power Plant Unit 2	1,065	BWR	OL 1974	Tennessee Valley Authority	1975
Decatur	Browns Ferry Nuclear Power Plant Unit 3	1,065	BWR	OL 1976	Tennessee Valley Authority	1977
Dothan	Joseph M. Farley Nuclear Plant Unit 1	804	BWR	OL 1977	Alabama Power Co.	1977
Dothan	Joseph M. Farley Nuclear Plant Unit 2	814	PWR	OL 1981	Alabama Power Co.	1981
Scottsboro	Bellefonte Nuclear Plant	1,235	PWR	CP 1974	Tennessee Valley Authority	1993
Scottsboro	Bellefonte Nuclear Plant Unit 2	1,235	PWR	CP 1974	Tennessee Valley Authority	1995
ARIZONA						
Wintersburg	Palo Verde Nuclear Generating Station Unit 1	1,304	PWR	OL 1984	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Nuclear Generating Station Unit 2	1,304	PWR	OL 1985	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Nuclear Generating Station Unit 3	1,304	PWR	CP 1976	Arizona Public Service Co.	1987
ARKANSAS						
Russelville	Arkansas Nuclear One Unit 1	836	PWR	OL 1974	Arkansas Power & Light Co.	1974
Russelville	Arkansas Nuclear One Unit 2	858	PWR	OL 1978	Arkansas Power & Light Co.	1980

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
CALIFORNIA						
San Clemente	San Onofre Nuclear Generating Station Unit 1	436	PWR	OL 1967	So. Calif. Ed. & San Diego Gas & Electric Co.	1968
San Clemente	San Onofre Nuclear Generating Station Unit 2	1,100	PWR	OL 1982	So. Calif. Ed. & San Diego Gas & Electric Co.	1983
San Clemente	San Onofre Nuclear Generating Station Unit 3	1,100	PWR	OL 1983	So. Calif. Ed. & San Diego Gas & Electric Co.	1984
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 1	1,084	PWR	OL 1984	Pacific Gas & Electric Co.	1985
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 2	1,106	PWR	OL 1985	Pacific Gas & Electric Co.	1986
Clay Station	Rancho Seco Nuclear Generating Station Unit 1	873	PWR	OL 1974	Sacramento Municipal Utility District	1975
COLORADO						
Platteville	Fort St. Vrain Nuclear Generating Station	330	HTGR	OL 1973	Public Service Co. of Colorado	1979
CONNECTICUT						
Haddam Neck	Haddam Neck Generating Station	555	PWR	OL 1967	Conn. Yankee Atomic Power Co.	1968
Waterford	Millstone Nuclear Power Station Unit 1	654	BWR	OL 1970	Northeast Nuclear Energy Co.	1971
Waterford	Millstone Nuclear Power Station Unit 2	864	PWR	OL 1975	Northeast Nuclear Energy Co.	1975
Waterford	Millstone Nuclear Power Station Unit 3	1,156	PWR	OL 1985	Northeast Nuclear Energy Co.	1986
FLORIDA						
Florida City	Turkey Point Station Unit 3	646	PWR	OL 1972	Florida Power & Light Co.	1972
Florida City	Turkey Point Station Unit 4	646	PWR	OL 1973	Florida Power & Light Co.	1973
Red Level	Crystal River Plant Unit 3	806	PWR	OL 1977	Florida Power Corp.	1977
Ft. Pierce	St. Lucie Plant Unit 1	817	PWR	OL 1976	Florida Power & Light Co.	1976
Ft. Pierce	St. Lucie Plant Unit 2	842	PWR	OL 1983	Florida Power & Light Co.	1983
GEORGIA						
Baxley	Edwin I. Hatch Plant Unit 1	757	BWR	OL 1974	Georgia Power Co.	1975
Baxley	Edwin I. Hatch Plant Unit 2	771	BWR	OL 1978	Georgia Power Co.	1979

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
GEORGIA (continued)						
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 1	1,100	PWR	CP 1974	Georgia Power Co.	1987
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 2	1,100	PWR	CP 1974	Georgia Power Co.	1988
ILLINOIS						
Morris	Dresden Nuclear Power Station Unit 2	772	BWR	OL 1969	Commonwealth Edison Co.	1970
Morris	Dresden Nuclear Power Station Unit 3	773	BWR	OL 1971	Commonwealth Edison Co.	1971
Zion	Zion Nuclear Plant Unit 1	1,040	PWR	OL 1973	Commonwealth Edison Co.	1973
Zion	Zion Nuclear Plant Unit 2	1,040	PWR	OL 1973	Commonwealth Edison Co.	1974
Cordova	Quad-Cities Station Unit 1	769	BWR	OL 1972	Comm. Ed. Co.-Iowa-Ill Gas & Elec. Co.	1973
Cordova	Quad-Cities Station Unit 2	769	BWR	OL 1972	Comm. Ed. Co.-Iowa-Ill Gas & Elec. Co.	1973
Seneca	LaSalle County Nuclear Station Unit 1	1,078	BWR	OL 1982	Commonwealth Edison Co.	1984
Seneca	LaSalle County Nuclear Station Unit 2	1,078	BWR	OL 1983	Commonwealth Edison Co.	1984
Byron	Byron Station Unit 1	1,120	PWR	OL 1984	Commonwealth Edison Co.	1985
Byron	Byron Station Unit 2	1,120	PWR	CP 1975	Commonwealth Edison Co.	1987
Braidwood	Braidwood Unit 1	1,120	PWR	CP 1975	Commonwealth Edison Co.	1987
Braidwood	Braidwood Unit 2	1,120	PWR	CP 1975	Commonwealth Edison Co.	1988
Clinton	Clinton Nuclear Power Plant Unit 1	950	BWR	CP 1976	Illinois Power Co.	1986
IOWA						
Pala	Duane Arnold Energy Center Unit 1	515	BWR	OL 1974	Iowa Elec. Power & Light Co.	1975
KANSAS						
Burlington	Wolf Creek	1,150	PWR	OL 1985	Kansas Gas & Elec. Co.	1985
LOUISIANA						
Taft	Waterford Steam Electric Station	1,151	PWR	OL 1984	Louisiana Power & Light Co.	1985

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
LOUISIANA (Continued)						
St. Francisville	River Bend Station Unit 1	934	BWR	OL 1985	Gulf States Utilities Co.	1986
MAINE						
Wiscasset	Maine Yankee Atomic Power	810	PWR	OL 1972	Maine Yankee Atomic Power Co.	1972
MARYLAND						
Lusby	Clavert Cliffs Nuclear Power Plant Unit 1	825	PWR	OL 1974	Baltimore Gas & Elec. Co.	1975
Lusby	Calvert Cliffs Nuclear Power Plant Unit 2	825	PWR	OL 1976	Baltimore Gas & Elec. Co.	1977
MASSACHUSETTS						
Rowe	Yankee Nuclear Power Station	175	PWR	OL 1960	Yankee Atomic Elec. Co.	1961
Plymouth	Pilgrim Station Unit 1	670	BWR	OL 1972	Boston Edison Co.	1972
MICHIGAN						
Big Rock Point	Big Rock Point Nuclear Plant	64	BWR	OL 1962	Consumers Power Co.	1963
South Haven	Palisades Nuclear Power Station	635	PWR	OL 1971	Consumers Power Co.	1971
Laguna Beach	Enrico Fermi Atomic Power Plant Unit 2	1,093	BWR	OL 1985	Detroit Power Co.	1986
Bridgman	Donald C. Cook Plant Unit 1	1,044	PWR	OL 1974	Indiana & Michigan Elec. Co.	1975
Bridgman	Donald C. Cook Plant Unit 2	1,082	PWR	OL 1977	Indiana & Michigan Elec. Co.	1978
Midland	Midland Nuclear Power Plant Unit 1	492	PWR	CP 1972	Consumers Power Co.	Indef.
Midland	Midland Nuclear Power Plant Unit 2	818	PWR	CP 1972	Consumers Power Co.	Indef.
MINNESOTA						
Monticello	Monticello Nuclear Generating Plant	525	BWR	OL 1970	Northern States Power Co.	1971
Red Wing	Prairie Island Nuclear Generating Plant Unit 1	503	PWR	OL 1973	Northern States Power Co.	1973
Red Wing	Prairie Island Nuclear Generating Plant Unit 2	500	PWR	OL 1974	Northern States Power Co.	1974

Site	Plant	Capacity (Net MWe)	Type	Status	Utility*	Commercial Operation
MISSISSIPPI						
Port Gibson	Grand Gulf Nuclear Station Unit 1	1,250	BWR	OL 1982	Mississippi Power & Light Co.	1985
Port Gibson	Grand Gulf Nuclear Station Unit 2	1,250	BWR	CP 1974	Mississippi Power & Light Co.	Indef.
MISSOURI						
Fulton	Callaway Plant Unit 1	1,188	PWR	OL 1984	Union Electric Co.	1985
NEBRASKA						
Fort Calhoun	Fort Calhoun Station Unit 1	478	PWR	OL 1973	Omaha Public Power District	1973
Brownville	Cooper Nuclear Station	764	BWR	OL 1974	Nebraska Public Power District	1974
NEW HAMPSHIRE						
Seabrook	Seabrook Nuclear Station Unit 1	1,198	PWR	CP 1976	Public Service of N.H.	1986
Seabrook	Seabrook Nuclear Station Unit 2	1,198	PWR	CP 1976	Public Service of N.H.	Indef.
NEW JERSEY						
Toms River	Oyster Creek Nuclear Power Plant Unit 1	620	BWR	OL 1969	GPU Nuclear Corp.	1969
Salem	Salem Nuclear Generating Station Unit 1	1,079	PWR	OL 1976	Public Service Elec. & Gas Co.	1977
Salem	Salem Nuclear Generating Station Unit 2	1,106	PWR	OL 1980	Public Service ELec. & Gas Co.	1981
Salem	Hope Creek Generating Station Unit 1	1,067	BWR	CP 1974	Public Service Elec. & Gas Co.	1986
NEW YORK						
Indian Point	Indian Point Station Unit 2	864	PWR	Ol 1973	Consolidated Edison Co.	1974
Indian Point	Indian Point Station Unit 3	891	PWR	OL 1975	Power Authority of the State of New York	1976
Scriba	Nine Mile Point Nuclear Unit 1	610	BWR	OL 1969	Niagara Mohawk Power Co.	1969
Scriba	Nine Mile Point Nuclear Unit 2	1,080	BWR	CP 1974	Niagara Mohawk Power Co.	1986
Ontario	R. E. Ginna Nuclear Power Plant Unit 1	470	PWR	OL 1969	Rochester Gas & Elec. Co.	1970

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
NEW YORK (continued)						
Brookhaven	Shoreham Nuclear Power Station	820	BWR	OL 1984	Long Island Lighting Co.	1986
Scriba	James A. FitzPatrick Nuclear Power Plant	810	BWR	OL 1974	Power Authority of the State of New York	1975
NORTH CAROLINA						
Southport	Brunswick Steam Electric Plant Unit 2	790	BWR	OL 1974	Carolina Power & Light Co.	1975
Southport	Brunswick Steam Electric Plant Unit 1	790	BWR	OL 1976	Carolina Power & Light Co.	1977
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 1	1,180	PWR	OL 1981	Duke Power Co.	1981
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 2	1,180	PWR	OL 1983	Duke Power Co.	1984
Bonsal	Shearon Harris Plant Unit 1	915	PWR	CP 1978	Carolina Power & Light Co.	1986
OHIO						
Oak Harbor	Davis-Besse Nuclear Power Station Unit 1	874	PWR	OL 1977	Toledo Edison-Cleveland Electric Illum. Co.	1977
Perry	Perry Nuclear Power Plant Unit 1	1,205	BWR	CP 1977	Toledo Edison-Cleveland Elec. Illum. Co.	1986
Perry	Perry Nuclear Power Plant Unit 2	1,205	BWR	CP 1977	Toledo Edison-Cleveland Elec. Illum. Co.	Indef.
OREGON						
Prescott	Trojan Nuclear Plant Unit 1	1,080	PWR	OL 1975	Portland General Elec. Co.	1976
PENNSYLVANIA						
Peach Bottom	Peach Bottom Atomic Power Station Unit 2	1,051	BWR	OL 1973	Philadelphia Elec. Co.	1974
Peach Bottom	Peach Bottom Atomic Power Station Unit 3	1,035	BWR	OL 1974	Philadelphia Elec. Co.	1974
Pottstown	Limerick Generating Station Unit 1	1,065	BWR	OL 1984	Philadelphia Elec. Co.	1986
Pottstown	Limerick Generating Station Unit 2	1,065	BWR	CP 1974	Philadelphia Elec. Co.	1990
Shippingport	Beaver Valley Power Station Unit 1	810	PWR	OL 1976	Duquesne Light Co. Ohio Edison Co.	1976

[1] Has license to load fuel, but restricted to 0.001 percent of power.

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
PENNSYLVANIA (continued)						
Shippingport	Beaver Valley Power Station Unit 2	852	PWR	CP 1974	Duquesne Light Co. Ohio Edison Co.	1987
Goldsboro	Three Mile Island Nuclear Station, Unit 1	776	PWR	OL 1974	GPU Nuclear Corp.	1974
Berwick	Susquehanna Steam Electric Station Unit 1	1,052	BWR	OL 1982	Pennsylvania Power & Light Co.	1983
Berwick	Susquehanna Steam Electric Station Unit 2	1,052	BWR	OL 1984	Pennsylvania Power & Light Co.	1985
SOUTH CAROLINA						
Hartsville	H. B. Robinson S.E. Plant Unit 2	665	PWR	OL 1970	Carolina Power & Light Co.	1971
Seneca	Oconee Nuclear Station Unit 1	860	PWR	OL 1973	Duke Power Co.	1973
Seneca	Oconee Nuclear Station Unit 2	860	PWR	OL 1973	Duke Power Co.	1974
Seneca	Oconee Nuclear Station Unit 3	860	PWR	OL 1974	Duke Power Co.	1974
Broad River	Virgil C. Summer Nuclear Station Unit 1	900	PWR	OL 1982	So. Carolina Elec. & Gas Co.	1984
Lake Wylie	Catawba Nuclear Station Unit 1	1,145	PWR	OL 1984	Duke Power Co.	1985
Lake Wylie	Catawba Nuclear Station Unit 2	1,145	PWR	CP 1975	Duke Power Co.	1986
TENNESSEE						
Daisy	Sequoyah Nuclear Power Plant Unit 1	1,128	PWR	OL 1980	Tennessee Valley Authority	1981
Daisy	Sequoyah Nuclear Power Plant Unit 2	1,148	PWR	OL 1981	Tennessee Valley Authority	1982
Spring City	Watts Bar Nuclear Plant Unit 1	1,165	PWR	CP 1973	Tennessee Valley Authority	1986
Spring City	Watts Bar Nuclear Plant Unite 2	1,165	PWR	CP 1973	Tennessee Valley Authority	1987
TEXAS						
Glen Rose	Comanche Peak Steam Electric Station Unit 1	1,150	PWR	CP 1974	Texas Utilities	1987
Glen Rose	Comanche Peak Steam Electric Station Unit 2	1,150	PWR	CP 1974	Texas Utilities	1988

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
TEXAS (continued)						
Bay City	South Texas Nuclear Project Unit 1	1,250	PWR	CP 1975	Houston Lighting & Power Co.	1987
Bay City	South Texas Nuclear Project Unit 2	1,250	PWR	CP 1975	Houston Lighting & Power Co.	1989
VERMONT						
Vernon	Vermont Yankee Generating Station	504	BWR	OL 1972	Vermont Yankee Nuclear Power Corp.	1972
VIRGINIA						
Gravel Neck	Surry Power Station Unit 1	775	PWR	OL 1972	Va. Electric & Power Co.	1972
Gravel Neck	Surry Power Station Unit 2	775	PWR	OL 1973	Va. Electric & Power Co.	1973
Mineral	North Anna Power Station Unit 1	865	PWR	OL 1976	Va. Electric & Power Co.	1978
Mineral	North Anna Power Station Unit 2	890	PWR	OL 1980	Va. Electric & Power Co.	1980
WASHINGTON						
Richland	WPPSS No. 1 (Hanford)	1,266	PWR	CP 1975	Wash. Public Power Supply System	Indef.
Richland	WPPSS No. 2 (Hanford)	1,103	BWR	OL 1983	Wash. Public Power Supply System	1984
Satsop	WPPSS No. 3	1,242	PWR	CP 1978	Wash. Public Power Supply System	Indef.
WISCONSIN						
LaCrosse	LaCrosse (Genoa) Nuclear Generating Station	48	BWR	OL 1967	Dairyland Power Coop.	1969
Two Creeks	Point Beach Nuclear Plant Unit 1	495	PWR	OL 1970	Wisconsin Michigan Power Co.	1970
Two Creeks	Point Beach Nuclear Plant Unit 2	495	PWR	OL 1971	Wisconsin Michigan Power Co.	1972
Kewaunee	Kewaunee Nuclear Power Plant	515	PWR	OL 1973	Wisconsin Public Svc. Corp.	1974

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